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SM-2 6000 & 12000 eKW NUCLEAR POWER PLANT PRIMARY SYSTEM DESIGN

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NUCLEAR POWER ENGINEERING DEPARTMENT
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Vol. No. 1 of 4
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SM-2 6000 & 12000 eKW

NUCLEAR POWER PLANT

PRIMARY SYSTEM DESIGN

E. F. Phelps, Project Engineer

Issued: June 1, 1961

Contract No. DA-44-192-ENG-7 with the
U.S. Army Engineer Research and Development Laboratories,
U. S. Army Corps of Engineers
Fort Belvoir, Virginia

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48	R. E. Williams	61	E. F. Phelps
49	P. V. Oby	62 - 70	NPED File

ABSTRACT

This report summarizes the development studies for an SM-2 Nuclear Power Plant primary system. The studies include the following: horizontal steam generator, safety and alarm system, decontamination methods, waste disposal, preassembly and packaging, vapor container, vertical steam generator and feedwater control system, pressurizer development and design studies, fuel handling, superheater, and instrumentation.

The design analysis for the main primary system and the primary auxiliary system is* ~~contained in this report and includes~~ calculations, sketches and descriptions.

The results obtained from the development studies and design analysis are included for a 6000 eKW plant design which utilizes a single 28 tMW reactor system, and for the preliminary design of a 12,000 eKW plant design which utilizes two identical 28 tMW reactor systems.

MASTER INDEX

VOLUME 1

TABLE OF CONTENTS*

* Separate indexes follow Parts A and B, including Lists of Figures and Drawings.

TABLE OF CONTENTS* (CONT'D)

* Separate indexes follow Parts A and B, including List of Figures and Drawings.

PART I - PROGRAM REVIEW

PART I - PROGRAM REVIEW

TABLE OF CONTENTS

TABLE OF CONTENTS (CONT'D)

1.0 INTRODUCTION

This report (four volumes) covers the design studies and analysis performed by Alco Products, Inc. for the SM-2 Final Design.

The SM-2 Nuclear Power Plant, Final Design, Phases I and II, was undertaken by Alco under Contract No. DA-44-192-ENG-7, issued by the U. S. Army Engineer Research and Development Laboratories (Corps of Engineers), Fort Belvoir, Virginia. Phase I work, beginning May 1959, covered Design and Development Studies and Tests and Phase II, beginning in June 1959, covered the Plant Design Analysis. Phases I and II were scheduled for completion by June 30, 1960; an extension was granted to September 9, 1960.

When Alco started work on Phases I and II of the SM-2 Final Design, all efforts were directed toward a single SM-2 capable of producing a minimum of 6 MW (net) electrical power. The components and systems to be developed under Phases I and II were those from the SM-2 preliminary design performed by Alco under Contract DA-44-009-ENG-3506, and reported in APAE No. 40, Vol. I** and II***. Phases I and II involve design of the entire nuclear power plant except the core and vessel.

In conjunction with the SM-2 plant design, the U. S. Atomic Energy Commission initiated development work in October 1958 under Contract No. AT(30-3)-326 for a reactor core and vessel capable of operating the plant for one year. Detail design information on the nuclear design, reactor vessel, and control rod drives has been developed and reported under this contract in APAE No. 69*.

Alco directed all design efforts towards completion of Phases I and II of Contract DA-44-192-ENG-7 up to April 11, 1960. At that time, the USAERDL Contract Officer notified Alco to terminate all work on the 6000 eKW power plant. A contractual modification was issued for a rescoped program, consisting of a conceptual design for an SM-2 12,000 eKW Power Complex, specific outline drawings, and primary system specifications, and specific items which were not included in the work scope for Phases I and II.

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* Hoover, H. L., project engineer, "SM-2 Core and Vessel Design Analysis," APAE No. 69, March 8, 1961.

** Alco Products, Inc., APPR-1B Preliminary Design Study of Steam Electric Conversion System, Contract No. DA-44-009-ENG-3506 APAE No. 40, Vol. 1, December 11, 1958.

*** Knighton, G. W., "Preliminary Design Study of SM-2 (APPR-1B) PWR-Steam Electric Station," APAE No. 40, Vol. II, June 1959.

The conceptual design for the SM-2 12,000 eKW Power Complex was completed on May 27, 1960 and reported in APAE No. 68*. Seven specifications for the primary system equipment have been prepared as separate items: primary piping, steam generator, pressurizer, primary pump, primary system insulation, primary system shield tank, and primary skid.

Volume 1 of this report covers the primary system design performed by Alco under Phases I and II of Contract DA-44-192-ENG-7 for the SM-2 6000 eKW plant and 12,000 eKW Power Complex. Volume 2 is a companion volume of design analysis calculations. Termination date for work on the 6000 eKW plant was April 11, 1960, and for work on the 12,000 eKW Power Complex, June 30, 1960.

* Hoover, H. L., project engineer, "Conceptual Design of SM-2 12,000 eKW Power Complex," APAE No. 68, May 27, 1960.

2.0 PHASE I DESIGN AND DEVELOPMENT STUDIES AND TESTS FOR THE SM-2 NUCLEAR POWER PLANT

The status of the work undertaken by Alco for design and development studies is outlined by reviewing a summary of each study. The studies follow.

2.1 HORIZONTAL STEAM GENERATOR[(2)] *

2.1.1 Summary

A preliminary design of a horizontal steam generator for SM-2 has been made. The chosen configuration for this design was a two-drum unit consisting of a heat exchanger unit and separator drum interconnected by integral riser and downcomer. For the given duty in the SM-2 system, this design has a reflux ratio of 9.9:1 at full load. The response of this steam generator to load transients was investigated with help of the analog computer, using a steam generator computer model similar to the model used for the transient analysis of the existing vertical steam generator design.

A comparison of the results of the analysis for the horizontal and the vertical design indicated the following advantages in favor of the horizontal design.

1. Due to slower decay of the pressure and steam flow rate transients during an increasing load change, more energy will be available during the important initial time period of the transient for maintaining the turbine speed under increased load conditions; therefore, the horizontal design would contribute to better overall plant transient response.

2. The level surge during transients is smaller.

3. The height of the unit is approximately one half of the height of the vertical design. This could be significant in the choice of the vapor container configuration.

Since the vertical steam generator design presently incorporated in the SM-2 design meets the steam pressure and flow demand of the turbine, the above advantages are offset by the following disadvantages.

1. The unit is more costly.

2. The weight is greater.

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* Reference citations refer to reference list at end of Part I.

3. The unit is longer and would require more space on the skid.

4. A decay heat removal system would be required due to insufficient thermal head to utilize the thermal cycling as available with a vertical unit.

Separation of the moisture in the steam is expected to be satisfactory. The outlet steam quality will be 99.976 percent for a 30 percent step increase in load at the 85 percent load level (from 70-100 percent load). Adequate level control can be achieved with a simple proportional controller. A controller with a low gain is desirable due to the potential flashing of the downcomer liquid during increasing load transients. The decay of pressure and steam flow rate during a step increase in load is found to be slower for a larger pipe volume, a larger riser volume, a higher pressure level, and a larger reflux ratio. The pressure and steam flow transient are not appreciably affected by a change in heat transfer surface, primary water volume, downcomer water volume, and metal volume.

The steam generator computer model has been made capable of indicating instability of the natural circulation (if present in the system), assuming that natural circulation exists at the start of the load change. A natural circulation stability criterion has been developed from the formula used in the computer model. No instability of this type has been found for the horizontal steam generator design. If natural circulation does not exist at the start of an increasing load transient, instability may result from the transient movement through the transition zone, where the natural circulation will be established. A rough estimate indicates that this transition may occur at the 30 percent load level. An analysis of this form of instability is desirable.

2.2 SIMPLIFICATION OF SAFETY AND ALARM SYSTEMS FOR THE SM-2[3]

2.2.1 Summary

The SM-2 scram and alarm system was investigated to determine which SM-1 scram parameters may be eliminated from the SM-2 control design, while improving reliability of the plant, and to determine the maximum scram set points of the scram parameters retained. The results show that the high pressure primary coolant scram can be eliminated if relief valves are provided. The high temperature steam scram should be eliminated, since the secondary system steam temperature does not reflect the primary coolant temperature. Since the high pressure steam scram is duplicated by the functions of the low coolant flow scram and low pressure scrams, it may be eliminated from the control system.

The startup behavior of the SM-2 was investigated using the maximum seven rod bank reactivity addition rates. It was found that in the event of a continuous rod withdrawal incident at startup, a ten second period rod motion stop is more than sufficient to protect the system from damage. Period delays are incorporated in the period scram instrumentation to protect from unnecessary scrams due to spurious readings. The peak power is determined as a function of the power trip level. In the event of failure of the control system during a startup incident, the core power will continue to rise until an inherent shutdown mechanism stops the excursions. The method of Golian is used to determine the magnitude of the power excursion.

The set points for the power and outlet temperature scrams were determined using burnout heat flux ratio criteria. The burnout ratio as a function of time after the loss of the primary pump was determined in order to establish the minimum setting for the low flow scram.

The problem of representing the response of the SM-2 to rod motion at power was investigated using the analog computer. The results demonstrate that period control may be removed from the scram control system when the SM-2 operating at or above a power level of 2 MW.

2.3 DECONTAMINATION METHODS [4]

2.3.1 Summary

Two basic methods of chemical decontamination proposed for the SM-2 are decontamination of the complete primary system and decontamination of a primary system component, the steam generator.

In full system decontamination, the core is removed and the decontamination solution circulated through the entire primary system. Full system decontamination will utilize as much of the primary system as feasible. A temporary reactor vessel cover will be used for the decontamination operation to simplify filling and venting. The only other equipment necessary for decontamination will be the fill and drain system.

In partial system decontamination, isolation of the specific component is required. In some cases, the method of isolation may be extremely critical since isolation of the component may have to be performed in a radiation area. Of the various methods of partial system decontamination investigated, three methods considered feasible are presented in this report:

a. Fixed tube bundle decontamination
b. Removable tube bundle decontamination
c. Isolation valve decontamination

The three methods differ only in the way the component is isolated. In all cases, the steam generator was chosen as the component to be decontaminated, since it has the greatest contaminated surface area in the primary system excluding the core, and it is the component most likely to require maintenance. The three steam generator decontamination methods require an external circulation system as well as a fill and drain system.

An evaluation of the full system and partial system methods of decontamination revealed that full system decontamination is the most advantageous. With a full system method, the entire primary system would be decontamined, thus eliminating the possibility of redistribution of activity. Other advantages include a minimum of external equipment, low overall plant activity levels, and higher decontamination solution flow rates at no additional costs for pumping and piping. The major disadvantage of full system decontamination is the increased volume of decontamination waste solution that must be processed.

2.4 WASTE DISPOSAL(5)

2.4.1 Summary

Waste volume and activity for all SM-2 plant wastes were estimated. Problems of normal and periodic radioactive plant wastes were investigated. For laboratory and decontamination wastes, samples were obtained or prepared for experimental evaluation. Evaluation included ion exchange and precipitation processing methods and corrosion testing.

Three possible waste disposal systems were considered in the final evaluation: dilution, ion exchange and dilution, and evaporation and dilution. Ion exchange and evaporation waste disposal systems were compared on the basis of feasibility and costs. Particular emphasis was given to the processing of normal and periodic wastes in a remote location.

2.4.1.1 Conclusions

The following conclusions were reached on the basis of work performed **under this program:**

1. **Operation of SM-2 will result in an estimated accumulation of 1040 gal/mo of primary coolant waste and 170 gal/mo of laboratory wastes. Approximately 740 gal of the primary coolant waste will be due to the continuous sampling of oxygen and hydrogen in the primary system. Activity of these wastes after 4-6 weeks decay will consist primarily of Co-60, -58, Fe-59, Mn-54, and Ta-182. Maximum long-lived activity was calculated** to be 7.3×10^{-3} μc/cc after approximately 15-20 yr of reactor

operation. Fission product levels at shutdown were assumed to observed SM-1, Core I levels and would be insignificant (except for Sr-90) for waste disposal purposes after 4-6 weeks decay.

2. Processing of SM-1 laboratory waste samples by cation exchange resulted in decontamination factors (DF) of more than 200 across the resin. These wastes were acidic (pH-2) and contained 1520 ppm total solids and 535 ppm dissolved solids as $CaCO_3$. Results reported at other installations are not as encouraging as those reported in the present study.

3. Wastes resulting from SM-2 primary system decontamination by the caustic permaganate-citrate combination solution method will constitute the largest fraction of periodic plant wastes. Approximately 12,000 gal of waste will result from such decontamination. Waste activity depends upon reactor history and the DF obtained. For a full system decontamination performed after 5 yr of SM-2 operation with DF of 20, the activity in the citrate solution was calculated to be 6.1 μc/cc.

4. A practical method of processing decontamination wastes is to mix the spent caustic permanganate and citrate solutions and either allow the mixture to stand for 8 weeks at room temperature or heat the mixture at 85°C for approximately 3 hr. The resultant reaction precipitates manganese dioxide as a scavenger of radioactivity and permits decantation of the remaining basic supernate for subsequent processing. However, considerable ammonia is produced by mixing the chemical decontamination solutions.

5. Corrosion results indicate that carbon steel can be used to contain mixed decontamination solution at temperatures up to 50°C. However stainless steel should be used when the temperature of the solution is at or near boiling.

6. Because the SM-2 location is unknown, a flexible waste disposal design is required. A system based on evaporation and dilution to discharge levels affords the most versatility, particularly where there is a restriction on dilution water or discharge limits.

7. Normal plant wastes may require initial processing prior to dilution to discharge tolerance (1×10^{-7} μc/cc). The feasibility of processing SM-2 plant wastes by ion exchange is uncertain. The exchange methods are not practical for processing decontamination wastes because of the high solids content.

8. Evaporation costs for processing SM-2 plant wastes are lower than ion exchange costs when dissolved solids or monthly waste output are high.

2.4.1.2 Recommendations

A waste disposal system based on evaporation and dilution to discharge levels should be included in the SM-2 design. Provisions should include the following:

1. Two 5000-gal storage tanks, one for chemical decontamination solution and one for water flush cycles.

2. A 2500-gal tank to permit decay of normal SM-2 plant wastes prior to final processing.

3. Facilities for processing or ultimate disposal of both the manganese dioxide slurry resulting from decontamination wastes and the residues from evaporation.

2.5 PREASSEMBLY AND PACKAGING[6]

2.5.1 Summary

This study describes, discusses and evaluates the preassembly and packaging of the SM-2 plant. Reference documents were Volumes I[7] and II[8] of APAE 40, covering the preliminary design study of the SM-2 steam electric station.

The overall packaging concept offers definite advantages for the plant in general, and the SM-2 Final Design is proceeding on the basis of incorporating the design features produced in this study for truck and rail shipments.

The results of the study showed the following disadvantages of a packaged plant:

1. the plant layout of skids required more space

2. skid structural members increased overall costs.

In general, these disadvantages were offset by the following advantages:

1. prepackaged units reduced overall erection time and costs

2. the number of component packages were reduced,

3. a higher degree of reliability is obtained due to shop testing

4. equipment can be dismantled and relocated

The study developed skid sizes which are shown on the summary table on Page 12. In general most of the skids meet shipping allowances from a weight and size standpoint. Special consideration must be given to the primary skid which is determined from a height and weight standpoint.

Skids No. 1 thru 5 represent the arrangement of the electrical equipment. The 4160 v switchgear and power transformers are located in an outdoor sub-station. Auxiliary generator equipment, motor control center and batteries will be located in the secondary systems building.

Skid No. 6 represents the plant control console. Basic instruments will be mounted in the console for automatic operation of the plant. The console will be common to the primary and secondary systems and located on the mezzanine level of the secondary system building. For reliability the plant instrumentation is designed on the basis of electric hydraulic controls for major servicrs.

Skids No. 7, 8 and 9 represent the equipment components of a turbine-generator-condenser unit. The condenser is designed structurally to support the turbo-generator. The turbine-generator will be located on a mezzanine level of the secondary system building. The condenser will be located directly below on floor level.

Three locations for turbine exhausts were considered: top, bottom and side. A secondary system building layout was made for each basic turbine exhaust, and it was concluded that the bottom exhaust offered a more suitable design for a packaging concept. The plant during maintenance periods.

The secondary system auxiliary equipment is represented by Skids 10 thru 15. Feedwater heaters and pumps are located on a common skid. Condensate pumps were deleted from the skid. Although vertical condensate pumps offered greater plant reliability than horizontal pumps, the condensate pumps were too long from a shipping standpoint for adaptability to a skid design.

SUMMARY TABULATION OF PREASSEMBLED SKIDS OR PACKAGES

Tag No.	Description	Sizes Width x ht. x lg.	Approx. Weight, lbs. (dry)
1	Transformers & Switchgear	7' 8" x 9' 9" x 27' 6"	25,200
2	4160 volt switchgear	7' 8" x 8' 6" x 22' 2"	30,600
3	Motor control center	3' 4" x 8' 6" x 22' 8"	27,600
4	Auxiliary Equipment Skid	6' 0" x 8' 6" x 16' 2".	9,000
5	Battery skid	6' 0" x 4' 1-1/2" x 14' 7"	10,000
6	Control console	10' 0" x 4' 0" x 16' 0"	10,000
7	Steam turbine (skid tag No. for identity only)	11' 0" x 10' 0" x 21' 0"	70,000
8	Generator (skid tag No. for identity only)	9' 6" x 11' 10" x 29' 0"	60,000
9	Condensor (skid tag No. for identity only)	9' 0" x 10' 0" x 22' 0"	84,000*
10	Feedwater heater & pump skid	9' 0" x 10' 0" x 22' 0"	50,000
11	Water treatment skid	9' 0" x 9' 4" x 25' 0"	20,000
12	Lube oil purifier	8' 6" 5' 5" x 11' 0"	8,100
13	Condensate storage tank	8' 0" O.D. 15' 0"	3,000
14	Condensate pumps (skid tag for identity only)	2' 0" x 2' 0" x 10' 0" each	2,000 (each)
15	Steam jet air ejectors (skid tag for identity only)	3' 0" x 4' 0" x 6' 0"	3,000
16	Primary Auxiliary equipment skid	9' 6" x 9' 6" x 22' 6"	29,000
17	Primary skid	11' 0" x 13' 2" x 22' 0"	90,000
18	Spent fuel tank	9' 0" O.D. x 20' 0"	5,000
19	Steam generator tube bundle & shell	4' 0" O.D. x 14' 6" long	50,000 (total)

* Plus approx. 25,000 lbs for structural steel req'd for condenser - turbine supporting design.

Regarding secondary system water treatment, it was concluded that plant reliability would be increased by a water treatment facility consisting of demineralizers rather than an evaporator type facility. Conclusions were based on water purity, ease of control, logistics and maintenance.

Skids 16 thru 19 inclusive represent primary systems equipment. The primary system skid is based on the use of a vertical steam generator and concrete-water primary shielding.

2.6 VAPOR CONTAINER (9)

2.6.1 Summary

A number of feasible vapor container concepts for the SM-2 plant design were developed using a steel vapor container shell of either spherical or cylindrical configuration. The following vapor container shells were considered:

1. Spheres (50, 46, 34 and 26 ft dia)
2. Single vertical cylinder (20 ft dia)
3. Two interconnected vertical cylinders (17 ft dia)
4. Single horizontal cylinder (20 ft dia)
5. Sphere with pressure suppression (26 ft dia)

Comparison of the vapor container concepts showed that the vertical cylinder of the smallest feasible diameter (20 ft) with the use of earth back-fill for secondary shielding and horizontal spent fuel transfer to a cylindrical steel spent fuel storage tank located outside the vapor container constitutes the best choice for the SM-2.

2.6.3 Design Pressure of Vapor Container

Design pressure of each of the vapor containers was determined on the basis of the internal pressure development in the vapor container after a maximum probable accident, which assumes rupture of the primary system during conditions of the plant.

2.6.4 Secondary Shielding

Secondary shielding required to protect plant personnel during a maximum credible accident is equivalent to a wall thickness of about 7 ft of ordinary concrete. Installation of a 7 ft thick concrete wall around the vapor container is expensive and time consuming. A less expensive and easier way to obtain the required secondary shielding consists of the use of earth at the site itself either in contact with the vapor container shell or separated from it by a concrete retainer wall. Use of an exclusion area around the vapor container to reduce the required thickness of the secondary shielding was investigated. It was found that such an exclusion area would have to be prohibitively large to gain a significant reduction in secondary shielding.

2.6.5 Spent Fuel Handling

Spent fuel storage containers considered in the concepts were the rectangular concrete pit and the cylindrical steel tank. Transfer of spent fuel to the spent fuel container can be accomplished by the method used in the SM-1 and the SM-1A. A new approach whereby spent fuel can be transferred horizontally to the spent fuel pit outside the vapor container has the advantage that the elevation of the bottom of the spent fuel pit no longer must be located below the elevation required for safe storage of the spent fuel elements. Both methods were considered in the vapor container concepts.

2.6.6 Missile Protection

Potential missiles can be divided into self-propelled and jet-propelled missiles. Large vessels, as the reactor vessel, pressurizer and steam generator, could become self-propelled missiles ~~through~~ a blowout of the vessel wall. However, these vessels can be tied down sufficiently to prevent takeoff of the vessel when such a blowout occurs. Calculations were made of potential jet-propelled missiles to determine impact velocities of jet-propelled missiles of different weights (5 to 50 lbs), and propelled by steam issuing from different sized orifices. Calculations indicate that none of the missiles considered will be able to penetrate the vapor container shell of the selected vapor container concept.

2.6.7 Cost Evaluation

Cost comparison of the considered vapor container concepts indicates that the smallest sphere (26 ft dia) and single vertical cylinder (20 ft dia) are the least expensive configurations. The multiple vessel concept is not economically competitive with the single vessel concept. Application of pressure suppression by use of additional equipment is not advantageous for the SM-2 concept.

2.7 VERTICAL STEAM GENERATOR AND FEEDWATER CONTROL SYSTEM

2.7.1 Summary

The ~~computer~~ model and potentiometer settings for the vertical steam generator ~~test~~ model have been developed as part of the data prediction part of the test program. The resulting analog runs will be compared with the actual test data to prove the validity of the analog computation techniques used in the design of the vertical steam generator for SM-2.

Calculations have been made to predict circulation instabilities for different positions of the value in the downcomer pipe. Comparison of these data with the test data will indicate the validity of the circulation stability criteria developed under the Horizontal Steam Generator Analysis program.

After completion of analog runs for various operating conditions, the prediction of moisture separation data can be started. The prediction of test data is about 70 percent complete.

2.8 DEVELOPMENT & DESIGN STUDIES, PRESSURIZER [10]

2.8.1 Summary

Comparisons to SM-1 plant performance had indicated that the computer model previously used to determine pressurizer performance and select the proper size was significantly in error in treating loss of load transients because it failed to account for the condensation which automatically takes place on the vessel wall and liquid interface. This also made the model incapable of evaluating methods of augmenting this condensation.

In the present work, the most appropriate condensation rates that could be arrived at from the literature were selected, and the analog computer model revised and greatly augmented to incorporate this phenomenon. Using this model, a generalized curve was derived for sizing a pressurizer for the SM-2 or similar PWR plants.

The new model was also used to evaluate some new design concepts, one of which appears very worthwhile for large pressurizers, or wherever size is particularly critical. This design uses a standpipe in the bottom of the pressurizer, forming an extension of the connecting pipe, to bring the insurge of primary loop water directly to the surface of the liquid. The results showed that this procedure will bring this relatively cool water in contact with the steam just as effectively as sprays. Both the standpipe and the sprays (fed from loop water) can produce overquenching, and must either be pressure-regulated or have an adjustable proportioning device which can be set during plant startup.

If automatic pressure-regulated sprays or standpipe are provided, the total space required between minimum and maximum pressures due to positive and negative load transients can be further reduced by making the target steady state pressure a function of steam generator load, with the no-load approximately 100 psi higher than the full load pressure.

2.9 FUEL HANDLING

2.9.1 Summary

The design program for the SM-2 fuel handling tools was started with the objective of designing tools that were functional, light in weight, easily handled, foolproof in operation, and low in cost. Tools were to be designed for handling the core components during the core loading or re-fueling procedure.

For handling operations that are repetitious and that require a high degree of maneuverability, a tool with a telescoping handle operated from one level would be most practical. It was decided to design a telescoping handle for use with removable and adaptors to handle multiple core components such as the fuel elements, absorbers and control rod caps.

A telescoping tool handle was designed utilizing a crank operated cable and pulley system for telescoping two tubes. Another cable system was used to operate the latch mechanism on the end adaptor. End adaptors were provided for handling the fuel elements and absorbers and for retrieving dropped objects.

This concept has several advantages over the SM-1 type of fuel handling tools. The telescoping tool does not require any counter-balance weights because the tool can be crane supported and the control handles can be kept at one operating level. The crank-operated telescoping feature provides the operator with more precise control of the tool.

Data supplied by the cable manufacturer indicated that the stretch in the latch cable might be too great for adequate operation of the latch mechanism.

Another telescoping tool handle was designed with a tube and screw operated latch system for more positive operation. This too, retains the same handling techniques and advantages as the original concept. End adaptors were designed to be used with this handle. These are the fuel element tool retrieving tool, and the control rod cap tool.

A tool is required to lift the core grid plate which weighs approximately 300 pounds. The telescoping handle is not adequate to lift this plate, so another handle was designed to be used for this purpose. An interchangeable end adaptor was designed to be used as the tool for attachment to the grid plate.

The grid plate latches are operated with another tool, which also is an interchangeable adaptor for the handle.

These seven fuel handling tools complete the list of tools designed for handling the SM-2 core components.

2.10 SUPERHEATER[11]

2.10.1 Summary

The superheat cycle with moisture separation offers the greatest overall advantage of all cycles studied. The advantages of the superheat cycle using 600 psia saturated steam are:

1. Increased turbine reliability by reduced moisture content of the steam during steady state and transient load conditions.

2. Smaller steam generator, resulting in greater ease of packaging.

3. Slight decrease in steam storage time lag in the turbine during load transient.

4. Lower capital cost.

The disadvantage of the superheat cycle, compared to the 600 psia saturated steam cycle, is its lower cycle efficiency, resulting in slightly increased fuel costs.

The advantages of the superheat cycle with external and initial moisture separation outweigh the disadvantages. Therefore, this cycle is the final recommendation for the SM-2 final design.

2.10.2 Thermal Cycle Efficiency

The reheat cycle with reheat provided by the steam generator results in a 2 percent increase in efficiency over the saturated steam cycle at 600 psia. The cycle with saturated steam at 600 psia and moisture separation shows an increase of 0.8 percent over the reference cycle. The cycle with superheat and moisture separation indicates an increase of 0.77 percent over the reference cycle. The use of superheat alone shows no increase over the reference cycle. The moisture separation mentioned above involves both external and internal moisture separation. The internal reheat cycle is the most efficient cycle at 29.95 percent efficiency.

2.10.3 Reliability

The turbine exhaust steam of the reference cycle with saturated steam at 600 psia can be reduced in moisture content by 11 percent with the use of reheat and by 5 percent with the use of superheat and moisture separation combined. These considerable reductions in moisure content will result in a considerable reduction in turbine blade erosion, which lowers turbine lifetime. Since the internal reheat cycle shows the greatest reduction in moisture content of the steam, this cycle offers the greatest reliability.

2.10.4 Ease of Packaging

The steam gnerator for the superheat cycle offers the greatest advantage with respect to ease of packaging as indicated by the following comparison of overall steam generator dimensions:

	ID	Length
600 psia saturated	52 in.	22 ft 6 in.
Superheat	49 in.	21 ft 6 in.
Reheat	54 in.	23 ft 6 in.

The turbine generator for the superheat cycle also offers the greatest advantage with respect to ease of packaging since it is 3 ft shorter than the unit for saturated steam at 600 psia and 5 ft shorter than the reheat unit.

2.10.5 Transient Response

The following tabulation summarizes data from analog runs on the frequency transient response problem.

SUMMARY OF DATA FROM ANALOG RUNS

Turn speed controller with integral and proportional gain of 15 and 40 respectively.

	Max. Frequency Deviation (%)	Time to Return Within Specifications (sec)
Frequency Specifications	0.83	1.5
A. Straight turbine without feedwater heaters	0.82	0.9
B. Turbine with feedwater heaters (min steam vol)	1.08	1.0
C. SM-2 design	1.13	3.5
D. SM-2 design with moisture separation	1.20	3.6
E. SM-2 design with moisture separation and reheat	1.40	16.0

Turbine speed controller with integral, proportional and derivative gain of 15, 40 and 15 respectively.

	Max. Frequency Deviation (%)	Time to Return Within Specifications (sec)
C. SM-2 design	0.69	1.3
D. SM-2 design with moisture separation	0.75	1.3
E. SM-2 design and moisture separation and reheat	0.99	2.0

A 13- stage Worthington steam turbine rated at 7500 K was used as the source of data for these analog runs. The gains shown above for the turbine speed controller in run 1 are based on a conventional droop governor with reset action. The gains for run No. 2 would require a special governor with fast action similar to electronic controllers.

In the first run only the straight turbine with no feedwater heaters meets the required specifications. Any additional equipment in the plant cycle caused both the maximum deviation and time to return to exceed the specification limits. In the second run the transient response specifications were met with the cycle with saturated steam at 600 psia and external and internal moisture separation. The superheat cycle with moisture separation would show a slightly better transient response since the time lag due to steam storage is inversely proportional to the specific volume of the steam. The specific volume of the steam of the superheat cycle is approximately 25 percent less than for the cycle with saturated steam at 600 psia. The reheat cycle presents the greatest problem in steam storage time lag and therefore would be the least desirable from the transient response standpoint.

2.10.6 Economics

The manufacturer's selling price of the steam generator design with superheat is $7,000 less than the steam generator design for saturated steam at 600 psia. Plant operating cost with the superheat design will increase slightly because of the reduced thermal cycle efficiency and resulting lower electrical output. From the SM-2 economic evaluation in APAE Memo No. 224(12), the increase in fuel costs for a reduction in electrical output would be 0.020 mils per kw hr or $1,316 annually. The reheat unit would cost approximately $260,000 more than the superheat unit, but most of this could be offset by decreased fuel costs.

2.11 INSTRUMENT STUDY

2.11.1 Summary

2.11.1.1 Requirements

An instrumentation study was made to derive information for use in the design of the instrumentation and control system for the SM-2 plant. The requirements of this study were as follows.

1. To determine if an instrumentation and control system can be designed and provided which will afford 100 percent reliability of instruments and 100 percent reliability of plant output.

2. To study past operating experience to determine which plant parameters should be instrumented, how they should be instrumented, and what type of readout should be provided.

3. To investigate the feasibility of using data logging methods such as magnetic tapes with digital or analog readout.

4. By using information obtained in the study to evolve a control philosophy which will make the best use of a centralized control system and increase plant efficiency.

2.11.1.2 Approval

Several steps were taken to meet the requirements of the instrumentation study. A study was made of SM-1 instrumentation. The men who had spent time operating the SM-1 plant were interviewed; a search was made of the operating reports submitted on SM-1. The designs of the instrumentation and control systems for the SM-1A and the PM-2A were reviewed to determine if and where weak points existed in the circuitry or design that would affect reliability. A literature search was made to be sure the most recent developments were used. Conferences were held with a total of twelve vendors in the field of instrumentation and control to determine the availability of systems using static and solid state components.

2.11.1.3 Conclusions

An instrumentation and control system can be designed and supplied that is 90 percent reliable. By use of such means as coincident circuitry and redundant instrument channels, a design could be devised that would approach 100 percent availability.

To insure the reliability of the system, solid state components would be used as far as possible. Only the primary sensing elements and the final readout and control instruments would have any moving parts.

Previous control room designs have over-emphasized the reactor instrumentation and control. Pressurized water reactors are inherently stable and need not be monitored constantly. The main product of any power plant is electricity. Consequently, it is felt that a shift in emphasis is desirable in regard to which parameters should be considered of prime importance when designing the console layout. For example, parameters such as reactor output and neutron flux would be considered of greater importance during normal operation than rod position indicators and would be positioned in the console layout to effect this degree of importance. Secondary system variables of major importance are items such as KW output, voltage, frequency, main steam pressure, and they should also be positioned according to importance.

A system of interchangeable plug-in recorders and indicators is desirable for situations in which a variable must be continuously monitored for a certain period. Trend recorders located at the console corners could also be used. Certain variables indicated at the console could then be switched to read out on the trend recorders if desired.

Studies indicate that data logging methods would not be feasible for use in an installation such as SM-2. Equipment is available which would do the job. However, there are several drawbacks:

1. The equipment is quite costly.

2. Reliability and availability would be adversely affected due to the complexity of the system and the fragile nature of some of its components.

3. Additional personnel would be required; these must be highly trained computer men.

3.0 PHASE II DESIGN ANALYSIS

3.1 PRIMARY SYSTEM

The design analysis for the primary system is described in detail in Part II of this volume.

3.2 SECONDARY SYSTEM

The design analysis for the secondary system is described in detail in volumes 3 and 4 of this report.

4.0 CONCEPTUAL DESIGN OF SM-2 12000 eKW POWER COMPLEX[1]

4.1 SUMMARY

The conceptual design describes a feasible plant capable of meeting the stringent requirements of operation and control necessary for Nike-Zeus application.

The basic plant consists of two SM-2 28 tMW pressurized light water reactor systems. These have stainless steel clad plate-type fuel elements and vertical steam generators (designed and developed under AEC Contract AT-(30-3)-326 and ERDL Contract DA-44-192-ENG-7,) in conjunction with five 4000 eKW steam turbine-generators. In addition to the basic plant, three oil-fired boilers are provided as standby sources of energy. Continuity of power generation during chage over from reactors to standby power is achieved by supplying the turbines with steam from accumulators for the estimated 3 min required to bring stand-by boilers to full steam flow.

Normal operation of the power complex is with four steam turbines operating at 75 percent of full rated load and steam supplied by the two reactor systems. Upon failure of any operating turbine, the full load demand can be supplied by the remaining turbines. The fifth turbine would then be put on the line and each of the four operating turbines would return to their 75 percent rated load operating condition to produce full plant load.

During refueling of a reactor or any unscheduled outage of a reactor system, the accumulators will supply steam until two boilers assume the steam demand.

In the light of a recent Alco study of a single turbine generator control, which included manufacturers' consultations, it is felt that a program consisting of a detailed simulation of the multiple turbine generator controls will indicate that the SM-2 12000 eKW Power Complex design can meet power quality requirements for the Nike-Zeus application.

This program will be conducted simultaneously with the final design and will dictate the necessary plant modifications, if any, to achieve the necessary power quality.

The design has in it many features deemed necessary to meet extremely high reliability and availability of the plant and its components. The major features added are:

1. Systems have been designed to produce the 12,000 eKW site requirement and the estimated 2000 eKW plant load in such manner that the reactor will operate at approximately 26 tMW rather than the core design load of 28 tMW.

2. Our experience over the past years with instrumentation has been. included so as to provide a high degree of reliability as is available.

3. The 23 degree superheat in the steam generated by the nuclear system steam generators in conjunction with external inter-stage moisture separation for turbines adds considerably higher reliability and increased life to the turbine.

4. The use of headered steam systems adds flexibility in operation, isolation of turbine units, and dampens hunting of slightly mi-matched units.

5. Generally every two operating pumps have one stand-by unit inter-locked with either of them to produce uninterrupted operation.

6. Multiple turbine arrangement produces an extremely flexible and reliable system.

7. The double-bus electrical system prevents failures of the complete plant caused by improper operation of any single circuit-breaker upon failure of a generator.

8. Control of the hot stand-by boilers is arranged to fire main burners upon alarms of various safety and operating sensing devices rather than by actual scram signal to minimize time to reach full steaming required for asssumption of load.

9. Necessary radiation deflection equipment is provided to prevent over-exposure to personnel.

10. By-passes around various items are provided to permit isolation for maintenance during operation.

11. A decontamination system is provided to permit decreased maintenance time over the life time of the plant.

12. The steam generator is designed to meet design conditions with 5 percent of tubes plugged.

13. Steam generator tube material type 304 SS is used, with which we have had the most successful experience in fabrication and operation for nuclear operation.

14. The nuclear system steam generator shell and tube bundle is removable without breaking or cutting the primary piping system.

Control of the plant is achieved through a single control console. Motor starters are mounted on the preassembled skids with remote controls at the console. Automatic interlocking upon failure of pumps is used to start-up the stand-by units. All indications and controls for normal operation at the console are located at that point.

Solid state instrumentation is used wherever possible for reliability.

Preassembly of components is included in the design. At present, 58 modules or separate components are anticipated. Maximum height and weight of any one component is 13 ft 3 in. high and 115, 000 lb. The maximum applies to a primary system skid.

A present estimate of the required operating crew is one plant superintendent, four shift supervisors, four instrument-operators, four mechanical technicians, two maintenance men and one process control man. This totals 16 men for 3-shift, full week coverage.

The fuel requirement for the plant will be two reactor cores per year and approximately 657, 000 gal of diesel oil, based on 400 hr of boiler operation and 8360 hr of standby boiler operation.

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PART II - DESIGN ANALYSIS

PART II - DESIGN ANALYSIS

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

VOLUME I

LIST OF DRAWINGS

LIST OF DRAWINGS (Cont'd)

LIST OF DRAWINGS (Cont'd)

STUDY PRINTS-NOT FOR CONSTRUCTION

1.0 SM-2 PLANT DESIGN, GENERAL

1.1 INTRODUCTION

This section covers the primary system design as associated with a 6000 or 12,000 eKW secondary system. In general, the primary reactor system and auxiliary equipment is designed to be compatible with either a 6000 or 12,000 eKW secondary system, the difference being that the 6000 eKW utilizes a single 28 tMW reactor and the 12,000 eKW utilizes two 28 tMW reactors. In either case the primary system design is identical for both plants except for some physical arrangement differences in the chemistry laboratories, demineralizer room and waste disposal systems.

The 6000 eKW plant design is based on the plant arrangement shown in APAE No. 40, Vol. II [1]; the 12,000 eKW plant design is based on the Conceptual Design of the SM-2 12,000 eKW Power Complex. [2]

The work performed under Phases I and II followed the major design criteria as required in Volumes I [3] and II [1] of APAE No. 40 covering the preliminary design study of the SM-2 PWR Steam Electric Station.

The main objective of the SM-2 program was the development of a PWR steam electric station capable of operating as a base load unit producing prime electrical power for all facilities comprising a Nike-Zeus acquisition radar installation. As failure of the plant would obviously be detrimental to the Nike-Zeus system, an extremely high degree of reliability had to be designed into the SM-2 plant.

In addition to the high degree of reliability required for the plant, the application of the SM-2 plant demanded that the power be generated with extremely small deviations in voltage and frequency. Discussions with steam generator manufacturers indicated that such stringent power quality requirements have never before been requested in any power plant. In addition, the SM-2 was designed to produce sufficient power to meet site requirements.

1.2 DESIGN CRITERIA

1.2.1 Reliability

The prime objective of the SM-2 plant was to produce quality electric power; therefore only proven and tested equipment was to be incorporated into the design. The generator should be capable of delivering continuous rated loads and the electrical characteristics should be such that at any load from zero to 100 percent of rated load, the voltage fluctuations should remain within plus or minus 1 percent of the rated 4160 v and frequency fluctuations within plus or minus 1/4 percent of the nominal 60 cps.

Upon any sudden application or reduction of 25 percent of rated load, the minimum frequency deviation from normal (as determined by oscillograph) should not exceed 1/2 cps. In addition, the frequency should return to and remain with the final steady state band by not more than 1-1/2 sec following the application or removal of load.

For a sudden load increase or decrease of 25 percent of the rated load, the voltage should not deviate more than 5 percent at the main power bus, and shall return to steady state tolerances within 1/2 sec.

1.2.2 Operation

The equipment is designed for reliable continuous operation with minimum personnel requirements (estimated).

6000 eKW Power Plant

One operator per shift
One equipment engineer per shift
One maintenance man (day shift-on call)
One instrument repair man (day shift-on call)
One process control man (day shift-on call)
One superintendent

12,000 eKW Power Plant

Four shift supervisors
Four instrument operators
Four mechanical technicians
Two maintenance men (day shift-on call)
One process control man
One superintendent

1.2.3 Maintenance

Accessibility to all pieces of equipment that require repair and replacement is to be provided. Storage facilities and work areas are included in building arrangements.

1.2.4 Packaging

The overall packaging design for the SM-2 offers definite advantages for the plant. It will:

1. Reduce overall installation cost

2. Permit pre-testing

3. Reduce overall erection time

4. Minimize the number of shipping packages

5. Allow for dismantling and relocation of equipment after a period of operation

6. Allow for some standardization if multiple units are considered

The plant will utilize approximately nineteen skids in the primary and secondary buildings. Shipping criteria are based on rail and truck limitations. Standard allowable skid sizes are 10 ft-8 in. wide by 10 ft-11 in. high by 32 ft-0 in. long; weight limitations are 66,000 lb per skid. In some cases where the limitations are exceeded, as the primary system skid, special shipment will be required.

The skids will be assembled in a centralized fabricator's shop, and the modules will be interconnected at the erection site.

1.2.5 Site Conditions

For the purpose of this work, the plant site has been assumed to be in the vicinity of Bismarck, North Dakota. Environmental conditions affecting plant design and construction are:

1. Altitude: 1650 ft above sea level

2. Temperature: Winter - high 21°F, average 10°F
   Summer - high 99°F, average 79°F

3. Wind loadings are assumed to be 20 lbs/ft^2 of projected area.

4. Soil bearing value is dependent upon local site material, but is assumed to be 4000 lbs/ft^2.

5. Seasonal frost is normally 6 ft.

6. Railroad and highway facilities are available at the site.

7. Labor in all classifications is available locally and no construction camp is required.

8. The cooling water for condensers is assumed to be lake water. The normal water level is 25 ft below maximum condenser height.

9. Construction crane facilities will be available.

10. Mobile crane will be available for removal of steam generator shell and bundle from vapor container as well as spent fuel cask from spent fuel pit.

1.3 OPERATING CONDITIONS

Primary System

Operating pressure, psia	2000
Reactor temperature	
Inlet ^{o}F	502.7
Outlet ^{o}F	527.7
Flow quantity, lbs/hr	3.05×10^{6}
Thermal output tMW	26.4
Blowdown	
Average lbs/hr	465
Maximum lbs/hr	930
Seal leakage, lbs/hr, approx.	50
Makeup	
Temperature ^{o}F, (approx.)	100
Flow, gpm	0-5

Secondary System

Operating pressure, psia	480
Operating temperature, ^{o}F	486
Feedwater inlet temperature, ^{o}F	339.3
Flow (from steam generator) lbs/hr	99.062
Electrical output, gross KW	7511
Blowdown, lbs/hr	310

1.4 ELECTRICAL REQUIREMENTS

1.4.1 Plant

Under the original concept of the SM-2 plant, the net electrical output required is 6 MW of electrical power. This output, added to the plant auxiliary load of approximately 750 KW, totals 6750 KW. To provide this output, the plant design calls for a 7500 KW turbine-driven generator operating at 4160 volts 3 phase 60 cycles, with steam supplied by the steam generator used with a single 28 tMW reactor.

The generator output is fed to a 4160 volt station bus to which are connected two 1000 KVA station service transofrmers and two 4160 volt distribution feeders. The two 1000 KVA transformers will relay each other in case of failure and thus insure continuity of service.

The power requirements of the SM-2 Complex III plant are a net output of 12 MW of electrical power. To provide this capacity, two SM-2 28 tMW reactors are provided operating in parallel to drive four turbine-driven electric generators. These generators are rated 5000 KVA, 4160 volts, 3 phase, 60 cycles. The total generator capacity needed with net output requirements of 12 MW and an auxiliary load of 1500 KW, is 13,500 KW. With four generators operating, each will be partially loaded. In the event of emergency shutdown of one generator, the three remaining generators will have sufficient capacity to carry the load. To further insure continuity of service, a fifth generator is provided for cold reserve.

These generators are connected to a double bus switching scheme which feeds duplicate station service transformers and two 4160 volt feeders. Either of the two station service transformers is capable of carrying the full station auxiliary load in the event of failure of the other. The double bus scheme minimizes the chance of station outage due to equipment failure or a bus fault.

1.4.2 Transient Response

A transient analysis of a pressurized water reactor electrical power plant was required because of stringent voltage and frequency specifications called out in Bell Telephone Laboratories Report G712071, issued December 1, 1958. These specifications are more stringent than those on any known power plant of this size, nuclear or fossil fuel fired.

A. Voltage Specifications

At any steady state load from zero to 100 percent of rated load, voltage fluctuation will remain within plus or minus 1 percent of the rated 4160 volts. For a sudden load increase or decrease of 25 percent of the rated load, the voltage shall not deviate more than 5 percent at the main power bus, and shall return to steady state tolerance within 1/2 sec.

B. Frequency Specifications

At any steady state load from zero to 100 percent of rated load, the frequency fluctuations will remain within plus or minus 1/4 percent of the nominal 60 cps.

Upon any sudden application or removal of 25 percent of the rated load, the maximum frequency deviation from normal shall not exceed 1/2 cps (0.83 percent). In addition, the frequency shall return to and remain within the final steady state band in not more than 1-1/2 sec following the application or removal of the load.

Electrical analogs were derived for the various systems comprising the plant. An electrical analog of a system consists of electronic amplifiers, passive elements such as resistors and capacitors, and non-linear devices, such as diodes and function generators. These electrical elements are connected together in such a way that voltages and/or currents replace the actual quantities of temperature, flow, pressure, etc., and behave in a manner similar to the quantities being simulated.

The success of the system studies in meeting the specified performance was found to be primarily a function of the voltage and the speed controllers.

An exciter response ratio of approximately 1.0 is required to meet the voltage specification. Since the voltage response depends on the generator type as well as the exciter system, an exact value of exciter response ratio is not specified. Exciter systems with response ratios of 1.0 are available on special order.

Isochronous type and proportional plus reset type speed controllers were found to satisfy the frequency specification. Analysis indicates that an isochronous speed controller is preferred. To obtain stable control with an isochronous speed controller, steam admission actuation lags must be held to a minimum. A proportional plus reset controller which meets the specifications with higher steam admission lags is available but is not a stock item.

Voltage control analysis can be executed with a higher degree of accuracy than can the speed control analysis. Test date should be obtained on the steam pipe and steam chest to validate the assumptions made for these effects on the speed control response. Preparations for these tests at SM-1 are now in progress.

Elimination of feedwater input to the steam generator analytical model shows that the feedwater system has no effect on the ability of the plant to satisfy the voltage or frequency specifications.

The specifications should not define a particular type of voltage or frequency control nor the control system parameters. This is because of turbine and generator design differences which affect the transient response. Manufacturers employ different methods to obtain the required voltage and frequency control, and advantage should be taken of these different methods.

A transient study of the systems proposed by vendors should be a required part of the vendor's proposal.

1.5 HAZARD REQUIREMENTS

Plant personnel and surroundings of the plant must be protected against harmful radiation emanating from the radioactive materials present in the plant, both during normal operation of the plant and during the maximum credible accident. Fission products accidentally released from the reactor system are to be contained in a vapor container, which will prevent the dispersion of activated materials into the countryside. The secondary shielding is to be designed to maintain the desired dose rates outside the shielding during the maximum credible accident.

The maximum credible accident consists of the simultaneous rupture of primary and secondary system within the vapor container and instantaneous release of the primary coolant and fluids contained in the steam generator. This rupture is assumed to occur during idling conditions of the plant, when the temperature of the fluids in the secondary system is equal to the average temperature of the primary system water. As a result of the exposure of the reactor core, meltdown of the core will occur and fission products will be released into the vapor container in the form of a cloud. The vapor container is to be designed to contain the released fluids and fission products for an indefinite time period after the occurrence of the maximum credible accident.

The secondary shield is to be designed to permit no more than 10 percent of laboratory tolerance, 0.75 mr/hr, exterior to the vapor container shield during normal full power operation. During the maximum credible accident, the secondary shielding shall be sufficient to limit the dose rate to 7.5 mr/hr outside a specified exclusion area. This exclusion area shall be such that there will be free access to the remainder of the power complex.

1.6 FOUNDATION AND STRUCTURES

The detail foundation design depends upon the character of the soil and its bearing value as determined by analysis of test borings made at the building site.

The foundation requirements of the structures follow conventional power plant loadings and conditions. A conservative approach has been made to minimize construction costs and time.

The structures follow the present concept of industrial building design for functional simplicity and ease of fabrication, erection and maintenance. No architectural decoration has been used. Grading is functional, and will provide both easy surface water drainage and earth shielding. Seismic forces have been evaluated in the preparation of the design.

1.7 REACTOR DESIGN

The SM-2 reactor is a pressurized water type with a thermal output of 28 MW. It is designed to operate at 2000 psia with approximate inlet and outlet water temperatures at 500°F and 532°F respectively and a coolant flow of 7800 gpm of light water. The flow is three-pass, with two passes upward through the core. The first pass is through the 24 peripheral fuel elements. The second is through the 14 central elements and 7 control rods. An intermediate pass is made downward through the reflector and thermal shield.

The cylindrical vessel, made of type 304 ss, stands about 14 ft high, has an ID of 41 in. and a wall thickness of 4-1/2 in. at the core position. It has a 2-7/8 in. thick hemispherical bottom head and a bolted top cover. The inlet and outlet nozzles, below and above the core region respectively, are of 11-1/2 in. ID and 18-1/8 in. and 17-1/2 in. OD. Penetration is also made in the vessel, below the core region for entrance of the 7 control rod drives. The internals of the vessel (core support, flow dividers, and the like) are of type 304 ss.

The reactor core is made up of 45 cells in a 7 by 7 array with corners removed. The 7 control rod elements contain 16 fuel plates, composed of Eu_2O_3 dispersed in a stainless steel matrix. The 38 fuel elements have 18 fuel plates, of highly enriched UO_2 with dispersed ZrB_2 as burnable poison. The plates are clad with type 347 ss.

1.8 REACTOR THERMAL AND HYDRAULIC DATA

Reactor thermal power	MW	28
Reactor power density	W/cc	200
Operating pressure	psia	2,000
Coolant inlet temperature	°F	500.5
Coolant outlet temperature	°F	525.5
Total mass flow	lbs/hr	2.7×10^6
Total volume flow	gpm	7,800
Number of passes through reactor		2
Flow area in first pass	ft^2	0.106
Flow area in second pass	ft^2	0.83
Coolant second pass velocity	fps (area 0.04 ft^2)	15.5 element
	(area 0.0353 ft^2)	17.5 cont. rod
Fraction of power in first pass		0.4048
Fraction of power in second pass		0.5952
Heat transfer surface, start of life	ft^2	600
Heat transfer surface, end of life	ft^2	732
Average heat flux, start of life	Btu/hr-ft^2	165,000
Average heat flux, end of life	Btu/hr-ft^2	135,000
Maximum surface temperature	°F	635.8

1.8 REACTOR THERMAL AND HYDRAULIC DATA (Cont'd)

Maximum metal temperature	^{o}F	684
Maximum coolant flow through fixed elements		
First pass	gpm	6,075
second pass	gpm	3,892
Coolant volume flow through control rods	gpm	1,967
Reactor frictional pressure drop	psi	6

1.9 REACTOR VESSEL DATA

Material	Forgings SA 336 FA type 304 SS	
	Plate SA 240 type 304 SS	
Inside diameter	in.	41.00
Wall thickness	in.	4.500
Design pressure	psia	2,200
Design stress	psi	11,600
Length (overall)	in.	154.25
Head thickness	in.	2.875
Head diameter	in.	55.75
Top opening diameter	in.	52.0
Cover design weight	lbs	4,850
Vessel design weight (incl. cover)	lbs	55,930

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2.0 SITE PLAN

The plot plan for the SM-2 6000 eKW (net) plant is shown on Dwg. M11594-6; the 12000 eKW plant is shown on Dwg. M11594-99.

2.1 BUILDINGS

2.1.1 Vapor Container

The vertical vapor containers used for either plant design are based on a maximum height of 35 ft-0 in. above grade due to site restrictions. The reactor is located at an elevation below grade in order to obtain the maximum effect of earth shielding. The use of a 20 ft-0 in. vertical vapor container was selected on an evaluated basis.[1]

Preliminary construction schedules were made in the SM-2 Preliminary Design indicating that the construction time required to complete the facility having the vapor containers adjacent to the secondary system wall would be reduced by approximately three months if the distances between the structures was 50 ft. This would maintain minimum construction interference, since it would preclude the use of extensive cofferdams which would be required for a tighter complex arrangement.

2.1.2 Demineralizer Room

Upon completion of the secondary systems building and vapor container complex, the excavated areas will be backfilled and an interconnecting area approximately 24 ft wide at grade level between structures will be used for the demineralizer room. The primary system auxiliary equipment will be located within the demineralizer room.

2.1.3 Secondary Systems Building

Building design for the station superstructure enclosing the power plant secondary system is entirely functional. A rigid type frame is favored with all features developed to permit adopting a prefabricated building.

2.2 OTHER FACTORS CONTRIBUTING TO PLANT LAYOUT

2.2.1 Cooling Water

The circulating water intake is located on the lake shore approximately 500 ft distant from the station.

2.2.2 Shielding

During normal operations, biological shielding of the primary system is provided in the demineralizer room and earth backfill around the vapor container. An exclusion area is established around the vapor container. Radiation levels outside the vapor container will not exceed 7.5 mr/hr or result in an expansion greater than 300 mr after rupture of the primary system for a 40 hr week.

2.2.3 Spent Fuel Pit

The spent fuel pit is to be located adjacent to the vapor container and kept filled with water.

2.2.4 Access Roads

Truck access to the spent fuel pit and primary demineralizer for shipping cask removal is required.

2.3 SITE ARRANGEMENT, 6000 eKW PLANT

This configuration (Dwg. M11594-6) represents the elementary power facility required to produce a minimum net output of 6000 eKW.

The plant consists of two main buildings and four smaller buildings. One structure is a prefabricated, steel frame unit 80 ft long and 50 ft wide by 35 ft high at the ridge, which houses the steam-electric conversion system, the central control room, the motor generator and battery room, boiler and feedwater skids, physics lab, repair area, office, locker room, toilet facilities and chlorinator room.

The other main structure is a steel, silo-shaped vapor container 20 ft in diameter projecting 35 ft above grade with the foundation submerged 30 ft below grade. (Ref. Dwgs. M11594-58 and M11594-56.) This cylindrical structure houses the reactor in a leak tight pressure vessel designed to retain all water, vapor and fission products released during a maximum credible accidental rupture of the primary system. This containing structure is blanketed with earth fill to act as secondary shielding required for biological radiation protection. The structure joining these two buildings encloses the interconnecting piping, the primary auxiliary skid, the fuel storage vault, the hot and cold labs, and emergency shower, and provides enclosed access between the reactor system and the steam-electric system. The 10-in. steam supply pipe running from the steam generator inside the vapor container to the steam turbine in the main building contributes a slight improvement to the transient response of the plant due to the added mass storage of steam.

Adjacent to the vapor container structure and installed below grade is a circular steel spent fuel tank, filled with water, designed to store 90 fuel elements and one shipping cask (Ref. Dwg. M11594-59). These two structures are joined by a fuel transfer pipe which is underground and below the water level of the spent fuel tank. The hot waste storage tanks are separated from the main buildings by a suitable distance required for shielding purposes. These storage tanks are enclosed in an underground concrete structure. Connected with this storage facility is a hot waste treatment and drumming station, also housed in an underground concrete building.

An auxiliary building on the lake shore houses the cooling water intake facilities. Cooling water is supplied to the plant via underground pipes from the lake shore pumping facility. Return water is also piped underground to the lake; a by-pass is provided to direct the warm discharge water towards the intake structure so as to provide some heating to the intake water during winter operation. Another building, 16 by 30 ft, houses chemicals. The electrical switch gear is the outdoor type, skid mounted, and is installed close to the main building.

Automobile parking is provided near the personnel entrances to the turbine-generator building. Truck access aprons serve all equipment and material handling doorways. A gravel access road to the cooling water intake structure is provided. Parking areas and driveways are paved with bituminous concrete.

A chain link fence encloses the areas of potential radioactivity to permit control of entry and exit.

The plant arrangement permits maximum flexibility from the operational and safety standpoint. Emphasis has been placed on accessibility to equipment with the turbine-generator room separate from the vapor container.

2.4 SITE ARRANGEMENT, 12,000 eKW POWER COMPLEX

This configuration (Dwg. M11594-100) represents a multiple primary and secondary combination to achieve maximum reliability of electric service where severe transient loads are anticipated. This arrangement of machinery offers a flexibility to withstand such loads and recover in a minimum time, returning to the steady state level of power output.

This power complex also represents an integrated facility for quality power generation. The system produces its steam from two reactor installations each of equal capacity (28 tMW) and of identical configuration, but each is independent and remote from the other and housed in separate identical vapor containers. Each has its own primary auxiliary systems and each is capable of either individual or joint operation. To back up the requirement of uninterrupted steam generation, three oil fired steam boilers are provided to operate as instant standby units; also, to support the constant steam supply, two large steam accumulators are connected in the main steam service line.

The main building which houses the steam-electric conversion system is a steel, rigid frame insulated panel, standardized unit fabircation, 176 ft long by 82 ft wide, housing the five steam turbine-generator units and related contributing auxiliary equipment and services with the grouping of steam, water and condensate essentials on one side while all electrical production, transformation, switching and control is arranged along the opposite side of the building. A traveling overhead crane serves the turbine-generator gallery. There are lift-out doors in the operating floor to allow the crane to reach items at the lower level where pumps, hot waste processing and other supporting facilities are located.

The vapor containers housing the primary systems are steel, silo-shaped pressure vessels of 20 ft dia. They project 35 ft above the grade with the foundation level 30 ft below grade. Earth is banked around to provide radiation shielding. These vapor container structures are located at such a distance from the main building so as to be in a safe zone from the radiation standpoint. The connecting building structure provides an area for the primary auxiliary equipment. Adjacent but outside the vapor containers are located separate spent fuel storage pits.

Electrical switchgear (4160 volts) is skid mounted and located outdoors; minimum length connections join it to the generators.

The cooling water intake structure is located about 500-ft from the main building on the lake shore and houses thepumps, trash screens, stop logs and other attendant equipment. Used water is returned to intake structure area to assist in tempering the lake water to retard freezing in cold weather.

Fuel oil tanks are installed on the side of the standby oil burning steam boilers. Access roads serve all areas and provide for personnel parking (Dwg. M11594-99).

2.5 OVERALL PLANT ARRANGEMENT

2.5.1 6000 eKW Plant

The SM-2 Plant for producing 6,000 eKW is arranged in three main areas; the vapor container, the turbine-generator building, and the interconnecting primary system auxiliary equipment and demineralizer room area (Dwgs. M11594-6 and M11594-7). The vapor container houses the primary skid, and the pressurizer. The primary auxiliary skid, fuel vault, and hot and cold labs are located in an area between the vapor container and the turbine-generator building. The turbine-generator building houses the secondary system, physics lab, chlorinator room, control room and equipment, battery and equipment room, and personnel facilities.

The turbine-generator building is a two story unit. The turbine and generator are located about centrally on the mezzanine floor. The condenser is parallel to and below the turbine-generator unit on the ground floor. The feedwater skid, the water treatment skid, and the lube oil purifier skid are also located on the ground floor. The latter two skids are arranged axially and are parallel to the condenser skid. Also parallel, but located on the opposite side of the condenser towards the primary auxiliary skid area is the feedwater skid. The chlorinator room and physics lab are found in adjacent corners of the building, on the ground floor, behind the water treatment and feedwater skids respectively. The condensate tank is placed on the roof of the chlorinator room.

Also on the fround floor, at the end of the building opposite the lab, and diagonally opposite the chlorinator room is the battery room containing the battery skid #5 and control and instrumentation power distribution skid #4. Adjacent to this is the office. The personnel facilities lie in the remaining corner. A repair area is also found on the ground floor on the outside wall adjacent to the lube oil purifier skid.

On the mezzanine, directly above the battery room is the 480 volt control center skid #3. Next to this, and facing towards the turbine, is the control console. These two units occupy the control room. Contiguous to the control room is the instrument repair room.

There are five other areas in the plant layout; the chemical storage area, the spent fuel pit, the circulating water intake structure, the waste treatment system, and the switchgear and power center areas. The latter consists of the 4160 volt switchgear and 480 volt power center and transformer. These lie parallel to each other outside the turbine-generator building with the former set adjacent to the wall near the water treatment and lube oil purifier skids. The spent fuel pit is located within the circular vapor container shielding area limits and is connected to the vapor container by the fuel transfer tube. The chemical storage area is separate from the main building and occupies an area near the switchgear and power center skids and the doorway entering the chlorinator room. The circulating water intake structure is located at the fresh water supply source and remote from the other areas mentioned.

The waste treatment system is composed of the waste storage area and the tank pit. The former, which lies behind the primary auxiliaries area, houses the waste storage drums, demineralizer system, waste evaporator, condenser and storage, storage flasks and compressor. The latter, which lies behind the storage area, houses the waste storage tanks.

2.5.2 12,000 eKW Plant

The SM-2 Complex III Plant for producing 12,000 eKW is T-shaped with two (2) vapor containers at the tips of the arms, the primary auxiliary areas forming the arms, and the main turbine-generator building forming the body. The layout is shown on Dwgs. M11594-100 and M11594-103. The vapor container houses the primary skid and is located along the wall near the vapor container in the arm of the T. The layout of the vapor container and the primary auxiliary area is the same in the two arms when facing out toward the vapor container. The turbine-generator building houses the secondary system, hot and cold chemistry labs, fuel vault, electrical load centers, flash evaporators, hot and cold condensate storage tanks, electrical equipment switchgear, waste disposal system, chlorine room, control room and equipment, personnel facilities and work and maintenance shops. The turbine-generator building is a two story unit. The system employs five turbine-generators, located on the station grade elevation and regimented with their axes parallel and corresponding assemblies in line. An area about the size of that taken up by a turbine generator is found between units #2 and #3 for maintenance. The control room, with the console and control center, abuts this area and lies on the outside wall approximately at the center. The five condensers are located below and parallel to the turbine-generator sets on the station basement elevation. The boiler feed pump package lies between condensers #2 and #3 below the space previously mentioned. The boiler feed booster and service water pump skid is located behind this package against the wall. There are three condensate pump packages located between condensers #1 & #2, #3 & #4, and #4 & 5 respectively. Three feedwater heater skids are located, one each at the exhaust end of condensers #1 & #2, #3 & #4, and #5. The two lube oil purifier skids lie on opposite sides of the boiler feed booster pump skid against the wall.

Behind turbine #4 are the two flash evaporators with mutually parallel axes at right angles to the turbine generators. In front of the evaporators are the cold condensate storage tank and the hot condensate storage tank in that order.

The electrical equipment 480 volt switchgear skid is located at the generator end of units #3, #4 and #5 with its axis perpendicular to the turbine-generator sets and lying perpendicular to the control center. On the same wall, to the left of the control room, facing the console, and contiguous with the control room and each other are the instrument repair room, health physics room, personnel facilities, and office space. The control and instrumentation power distribution skid and the battery skid lie directly below the control room. The two 4160 volt switchgear skids and the transformer skid falling parallel to each other are outside the main building behind the control room.

The remaining assigned areas within the building on the grade elevation contain the cold chemistry lab at the corner adjacent to the left arm of the "T" and the hot chemistry lab at the corner adjacent to the right arm, the fuel vault adjoining the hot lab and the back wall, and the chlorine room diagonally opposite the hot lab. The burner control stations for the three parallel boilers,found outside the building and at right angles to it, lie on the wall between the chlorine room and the hot condensate storage tank. Two parallel accumulator banks are also outside the building parallel with the boilers and perpendicular to the building at about the hot and cold condensate storage tanks.

The remaining assigned areas within the building at the basement elevation house the waste disposal processing equipment skids #1, #2 & #3 with the air compressor, and the air receiver, all at the head end of the "T", and the two storage areas at the corners with the maintenance workshop between at the base of the "T". The decay tank and hot waste tanks #1 & #2 are underground outside the building behind the waste disposal system.

Two remaining stations outside of the buildings are the spent fuel pit, which is located within the vapor container shielding area limits, and the circulating water intake structure located at the fresh water supply source.

2. 5. 3 Comparison of 6,000 eKW and 12,000 eKW Plants

There is considerable difference in the plant arrangements of the 6,000 and 12,000 eKW units; however there are some similarities. The vapor container - spent fuel pit area is laid out similarly. The primary auxiliaries section is similar except that the hot and cold labs and fuel vault are located in the main building on the larger unit. The 12,000 eKW unit uses five turbine-generator sets with five condensers and two lube oil skids. The 6,000 eKW unit uses one of each. The larger unit uses flash evaporators and storage tanks in place of the water treatment skid. Three feedwater skids and a boiler feed pump package are used on the 12,000 eKW unit. One feedwater skid is found on the smaller unit. The 12,000 eKW unit uses three auxiliary boilers and two accumulator banks while the smaller unit has none of these. The final major difference in the two arrangements is the inclusion within the building of the waste disposal system and storage facilities on the larger unit.

Reference to Dwg. M11594-100 and M11594-103 for the 12,000 eKW unit and Dwg. M11594-7 and M11594-8 for the 6,000 eKW unit will show more clearly the variances in plant arrangements.

2. 6 EQUIPMENT ARRANGEMENT

2. 6. 1 Vapor Container

The primary equipment of the power complex is located within the steel vapor container enclosure (Dwg. M11594-58). The skid mounted reactor, steam generator, lower shield tank, control rods, primary piping and primary coolant pump form the basic elements of the containment and are located on the bottom level well anchored to a concrete base. The lower shield tank supports the weight of the upper shield tank and its water. Concrete and lead shielding are mounted around the outside of the lower and upper shield tank. The pressurizer is mounted on a steel structural base which is secured to the concrete foundation with anchor bolts.

Since the pressurizer and its base are not part of the basic skid assembly, the piping to the primary coolant circuit is through field welded connections. Control rod drives and electrical devices are mounted to the basic skid assembly as a field installation. The pressurizer heating elements also are wired up to the control box as an electrical field installation. A sump pump and other minor devices are also located at this lower level.

The upper or operating level is at the plant ground floor level and consists of steel gratings mounted on a steel structure that is all bolt connected for ease of removal. Provision has been made to store the reactor dry cap in a suitable rack at this level. An air conditioner unit is wall mounted.

The upper shield tank is covered with removable checker plate lids and a movable service bridge is provided for the operating personnel. Reactor tools are used in conjunction with an overhead polar crane. The fuel transfer car and carriage is actuated by a steel tape contained in a reel which is wall mounted. The fuel transfer tube isolation valve has a remote mounted indicating handle located at this operating deck level.

Access to lower levels is by steel ship ladders.

2.6.2 Equipment Access and Manways

The primary system major components are enclosed within the vapor container vessel with the auxiliary equipment adjacent but beyond the secondary shielding (Dwg. M11594-58). Access for maintenance is through a container vessel code door (Dwg. M11594-60) at ground floor level. Crane access is through a 7 ft diam quick opening shear type closure (Dwg. M11594-60) overhead in the hemispherical roof segment of the vapor container, centered directly over the steam generator. The inside service crane is of the polar type. Penetrations for services are sealed and piping is anchored to prevent stress transfer to the container shell. Removable catwalks and hand rails are bolted to structural members, all of which may be disassembled to provide any clearance desired for service or access at lower levels. Vent connections are provided to exhaust the area through a filter system installed on the auxiliary skid. When ventilation is in progress, fresh air may be introduced through the opening of the code personnel door.

2.6.3 Demineralizer Room Arrangement

The primary system demineralizers are located on the primary auxiliary skid. The skid is placed in the demineralizer room (Dwg. M11594-6) to facilitate removal of the spent demineralizer in its shipping cask and replacement by a new one. Because of the weight and the hot condition of the spent demineralizer, the changeover should be accomplished rapidly to minimize the exposure time to personnel. To facilitate this operation, the room has been arranged to provide for a truck to enter and be loaded. A fork lift truck would be used for the loading and unloading operation.

Other primary system auxiliaries which are mounted on this skid are: blower and filter bank for the vapor container ventilation system, primary makeup pumps, spent fuel pit recirculating pump, vapor container cooling water pumps, primary makeup tank, vapor container cooling water tank, boron tank, primary and secondary blowdown coolers, spent fuel pit water cooler, various regulating valves, electrical controls, and devices for the operation and control of the mechanical items incorporated.

The skid is so placed in the area that all connecting piping can be racked against the wall, thus allowing easy access to all components.

Located at the vapor container end of the room is a pipe pit where the piping is grouped in a special vapor container penetration panel which is so designed that thermal stresses are prevented from being transmitted to the shell structure of the vapor container. A similar electrical penetration panel is provided. The main 10 in. steam line from the steam generator installed inside the vapor container passes through the demineralizer room to the main steam supply system. Electrical cables are also directed through this room nested in cable trays and conduits thus becoming the main link between the primary system and the steam-electric secondary system.

During normal operations, biological shielding of the primary system in the demineralizer room is provided by use of concrete and partially by earth backfill as shown on Dwg. M11594-56. The vertical walls are one foot thick and the ceiling about 1-1/2 ft thick. Radiation from the demineralizers is reduced locally by keeping the demineralizers in the shipping casks at all times including changeover of spent units with fresh assemblies. The primary makeup tank, which is located on the primary auxiliary skid in the demineralizer room, is not normally shielded; however, the tank is positioned vertically to permit stacking of lead bricks on the skid and around the tank should conditions warrant it.

The fuel vault, a safe storage area, is located in the demineralizer room. The vault is an enclosure formed by 12 in. thick concrete walls with racks for the stored fuel elements. The door to the vault, 3 ft wide, offers a rugged design for safe storage against fire and theft.

Space has been provided in the demineralizer room for adjoining hot and cold laboratories. Each laboratory will have floor and wall cabinets for the storage of analytical equipment, chemicals, glassware, books, and records. Hoods, sinks, and benches are also provided for the work being conducted in the laboratories. Primary water samples will be taken directly from the purification loop instead of running sample lines to the hot laboratory. This is being done to limit as much as possible the hot areas in the primary building. However, all secondary sample lines will come into the cold laboratory to facilitate laboratory work.

A standard safety shower is also located in the demineralizer room.

2.7 PIPING AND WIRING PHILOSOPHY

A packaged plant design was chosen to make use of the following advantages:

1. Reduced erection time and costs.
2. Higher degree of reliability due to shop testing.
3. Equipment can be dismantled and relocated.

Pipe and fittings for the primary system including mains and purification system are T304 ss with 250 microinch r.m.s. finish. Cooling water piping and other lines not in contact with primary water are of conventional materials suited to the particular service. To provide integrity and zero leakage, all piping is welded, with a minimum of flanges, and generally is of greater wall thickness than is necessary for pressure containment, particularly in the small sizes. Primarily this heavier pipe is used for mechanical strength, rather than for added corrosion allowance.

Interconnecting piping, field installed, follows the same pattern of materials and fabrication, providing for anchorage in a manner to avoid introducing thermal stress or pipeline thrust into the vapor container, while maintaining the integrity of that pressure vessel. Where necessary, piping and certain equipment is spring supported and contoured generally to provide a geometry which will allow the absorption of thermal stresses by the pipe itself.

The wiring was planned to contribute to the concept of shop assembly and testing, simple dismantling and shipment, and ease and cheapness of erection in the field.

Power cables will be in accordance with the National Electrical Code to provide ample current carrying capacity and to eliminate the possibility of problems caused by excessive voltage drop.

The voltage level of the control signal carried by control cables will be of such magnitude as to minimize problems caused by interference.

2.8 STORAGE FACILITIES

2.8.1 6,000 eKW Plant

In addition to the normal operating storage facilities such as fuel vault and waste storage tanks, certain areas have been provided for storage of equipment and materials necessary for plant maintenance and supply.

For generator maintenance and in-plant work, a tool and parts storage space approximately 25 ft by 2-1/2 ft is provided on the ground floor in an area abutting the primary auxiliaries and lab area and contiguous with the motor-generator and battery room. On the same floor, in the repair area, facilities are available for small parts torage. For instrument repair, storage bins are provided in the instrument repair shop. A chemical storage area 16 ft by 30 ft is provided outside of the main buildings. Storage for chlorine cylinders is provided along two opposite walls of the chlorinator room.

2.8.2 12,000 eKW Plant

In this plant, two general storage areas, each about 16 ft by 23 ft, are found in adjacent corners on the basement elevation on opposite sides of the maintenance workshop area. The work shop area, as well as the instrument repair room, have facilities for storing parts for immediate needs. Chlorine bottles will be stored in the chlorinator room. This plant has no chemical storage area. A fuel oil storage site for the auxiliary boilers is made up of two 40,000 bbl capacity tanks set on concrete slabs at site grade. These supply a 500 gal elevated storage in the power plant. Normal operating storage facilities such as fuel vault and condensate storage are described elsewhere.

2.9 WASTE DISPOSAL

Contaminated wastes, collected during normal plant operation or resulting from a decontamination program, will be stored and processed in two concrete enclosures apart from the main building (Dwg. M03-M1). They will be recessed in the ground to permit gravity flow of wastes from the plant drains to the 2500 gal receiving tank, and to reduce the shielding costs by utilization of soil as an attenuator.

The enclosure nearest to the main building is designated as the waste treatment building. It is divided into three areas by partitions which serve as shielding around hot equipment. The partitioned areas, identified by the process functions performed within them, are the evaporator room, drummed waste storage room, and the miscellaneous equipment room.

The evaporator room houses the evaporator-condenser and its supporting equipment including surge tank, pumps, flow meters, valves, piping, and the like. This equipment is preassembled, skid mounted, and transported to the site as one package.

The drummed waste storage room houses the 200 gal holding tank, one demineralizer with its shipping cask, one waste filter element shipping cask, and twenty-five concentrated waste storage drums. The filling process for the waste storage drums is also performed in this room.

The miscellaneous equipment room fills multiple requirements. It is a low radiation area and consequently serves as that area into which entrance is made by personnel when wastes are being processed. Entrance to the other process areas is also made from this room. The gaseous waste storage facilities are also contained here, and consist of one air compressor, one blower, and six storage flasks. A 300 gal distilled water tank, which collects and stores process by-product water of low activity, is located in this room near the entrance to the evaporator room. Finall, a valve panel is provided near the stairway from which process and utility fluids are controlled.

A monorail crane is located within an unheated shed above the waste treatment building. Its function is to deposit and remove waste storage drums from the drummed waste storage room, and to remove the evaporator when servicing is required. Hatches through the roof of the waste treatment building are provided to accomplish these operations.

Unused waste storage drums are stored outside the building. Filled drums, however, are stored in the drummed waste storage room until they can be shipped to the Government's waste processing station. Provision is made for storage of 25 20-gal capacity waste storage drums, based on the anticipated collection of approximately 500 gal of concentrated wastes accumulated as the result of a decontamination program. All drums and casks are to be shielded with sufficient lead or concrete to meet shipping regulations. Dollys are to be provided for use with the waste storage drums to facilitate easy maneuvering of the drums from below the hatch to the filling station, storage area, and back to the area below the hatch.

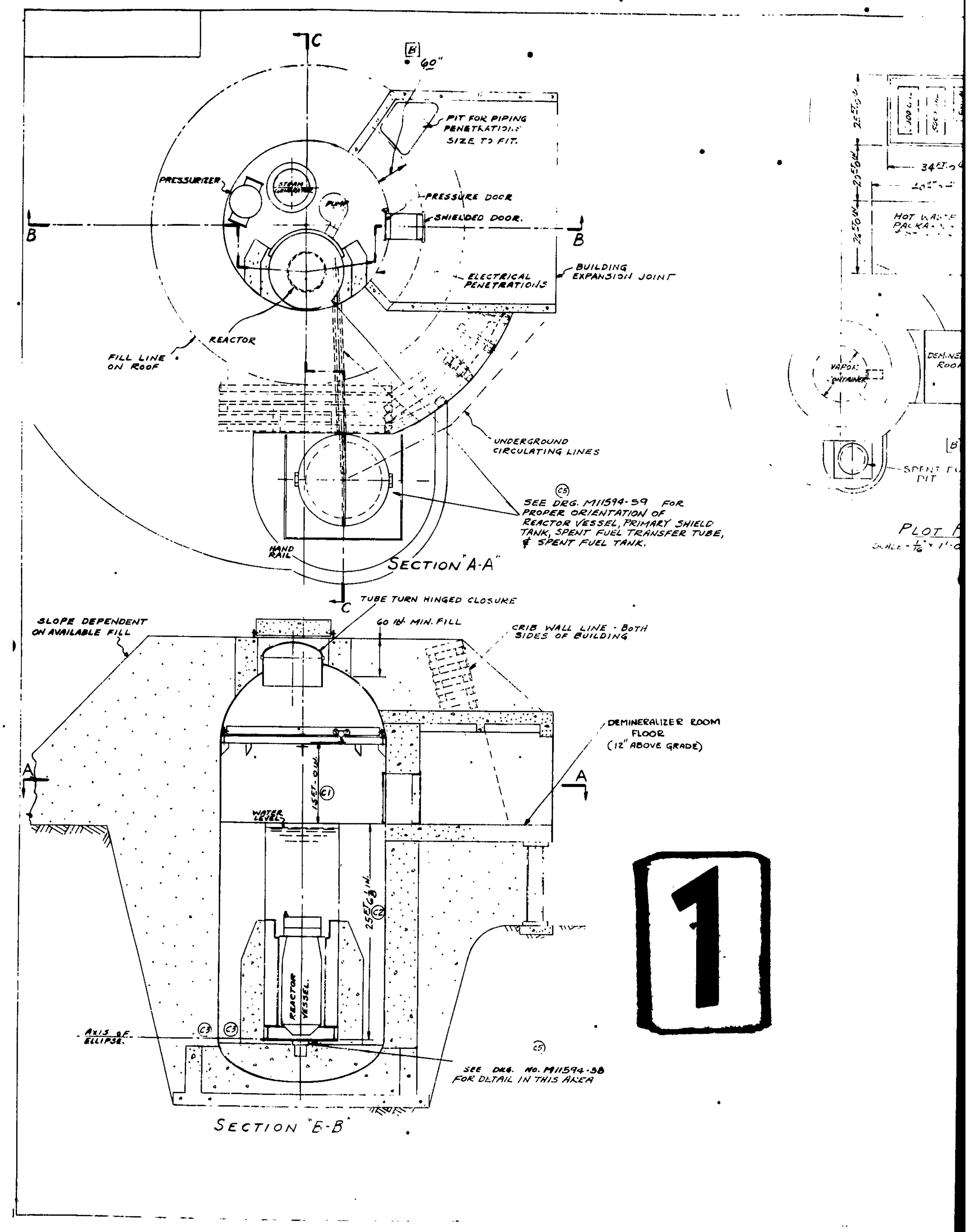
The hot waste storage building is essentially a concrete lined pit which houses two 5000-gal storage tanks, and one 2500-gal receiving tank. Motors and controls for pumping and mixing equipment on each tank are located above the 2-1/2-ft thick concrete slat roof to provide access for servicing. An unheated shelter is to be provided for this equipment. A hatchway through the covering slab and a ladder will be provided for entrance into the pit.

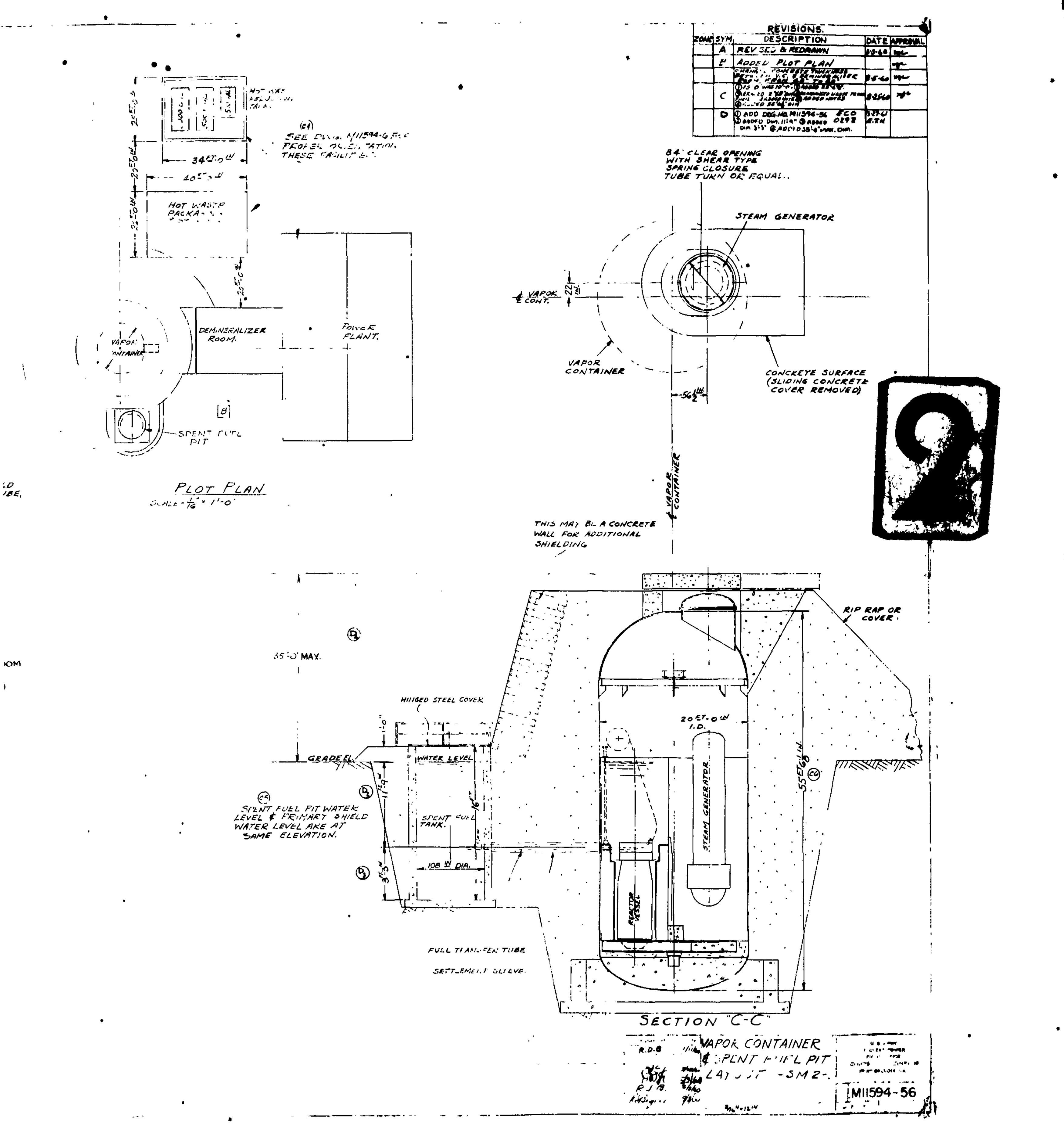
Pipe enclosures for process and utilities piping run from the main building to the waste treatment building, and then to the waste storage building.

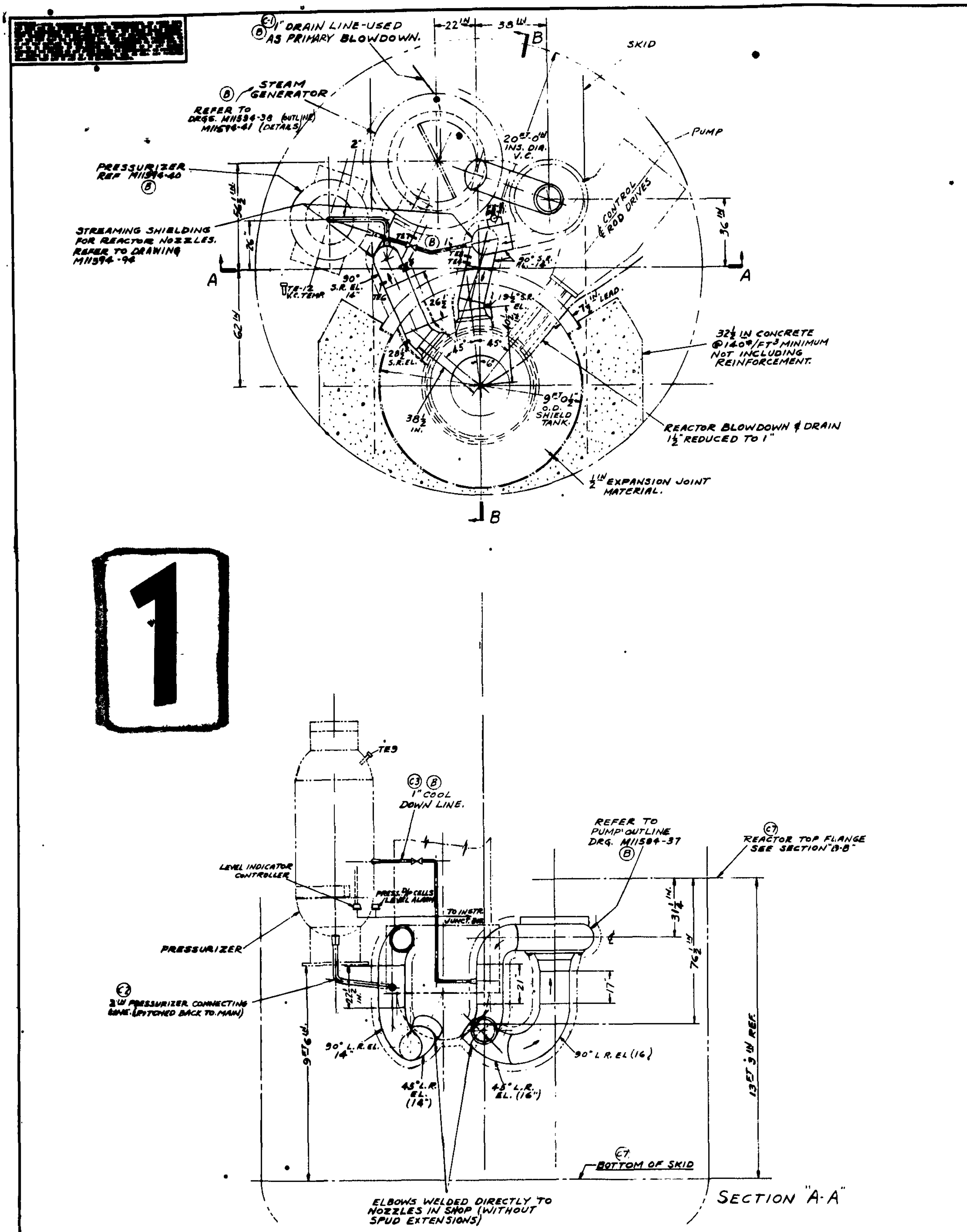
The waste storage building, waste treatment building, and pipe enclosures will be steam heated with steam available from the main plant.

REFERENCES

1. Van Kessel, H. F., "Vapor Container Concepts Study for SM-2," APAE No. 63, May 17, 1960.


C
B
60"
PIT FOR PIPING PENETRATIONS SIZE TO FIT.
PRESSURIZER
STEAM GENERATOR
PUMP
PRESSURE DOOR
SHIELDED DOOR.
B
B
ELECTRICAL PENETRATIONS
BUILDING EXPANSION JOINT
REACTOR
FILL LINE ON ROOF
UNDERGROUND CIRCULATING LINES
C5
SEE DRG. M11594-59 FOR PROPER ORIENTATION OF REACTOR VESSEL, PRIMARY SHIELD TANK, SPENT FUEL TRANSFER TUBE, & SPENT FUEL TANK.
HAND RAIL
SECTION "A-A"
C
TUBE TURN HINGED CLOSURE
SLOPE DEPENDENT ON AVAILABLE FILL
60 IN. MIN. FILL
CRIB WALL LINE - BOTH SIDES OF BUILDING
DEMINERALIZER ROOM FLOOR (12" ABOVE GRADE)
A
A
15FT.-0IN.
C1
WATER LEVEL
25FT.-6IN.
C2
REACTOR VESSEL
AXIS OF ELLIPSE
C3
C3
C5
SEE DRG. NO. M11594-58 FOR DETAIL IN THIS AREA
SECTION "B-B"
VAPOR CONTAINER
SPENT FUEL PIT
PLOT P
1


REVISIONS.
ZONE SYM. DESCRIPTION DATE APPROVAL
A REVISED & REDRAWN
B ADDED PLOT PLAN
C
D ① ADD DWG. NO. M11594-56
84" CLEAR OPENING WITH SHEAR TYPE SPRING CLOSURE TUBE TURN OR EQUAL.
STEAM GENERATOR
℄ VAPOR CONT.
22 IN
VAPOR CONTAINER
CONCRETE SURFACE (SLIDING CONCRETE COVER REMOVED)
56½ IN
℄ VAPOR CONTAINER
HOT WASTE PACKAGING
34 FT-0 IN
40 FT-0 IN
25 FT-0 IN
20 FT-0 IN
SEE DWG. M11594-6 FOR PROPER ORIENTATION THESE FACILITIES.
DEMINERALIZER ROOM.
POWER PLANT.
VAPOR CONTAINER
SPENT FUEL PIT
PLOT PLAN
SCALE - 1/16" = 1'-0"
THIS MAY BE A CONCRETE WALL FOR ADDITIONAL SHIELDING
RIP RAP OR COVER.
35'-0" MAX.
HINGED STEEL COVER
GRADE EL.
WATER LEVEL
SPENT FUEL TANK.
108 IN DIA.
11 FT-9 IN
3 FT-3 IN
16 FT
SPENT FUEL PIT WATER LEVEL & PRIMARY SHIELD WATER LEVEL ARE AT SAME ELEVATION.
20 FT-0 IN I.D.
STEAM GENERATOR
REACTOR VESSEL
55 FT-6 IN
FUEL TRANSFER TUBE
SETTLEMENT SLEEVE.
SECTION "C-C"
VAPOR CONTAINER & SPENT FUEL PIT LAYOUT - SM2-
R.D.B.
R.J.B.
M11594-56

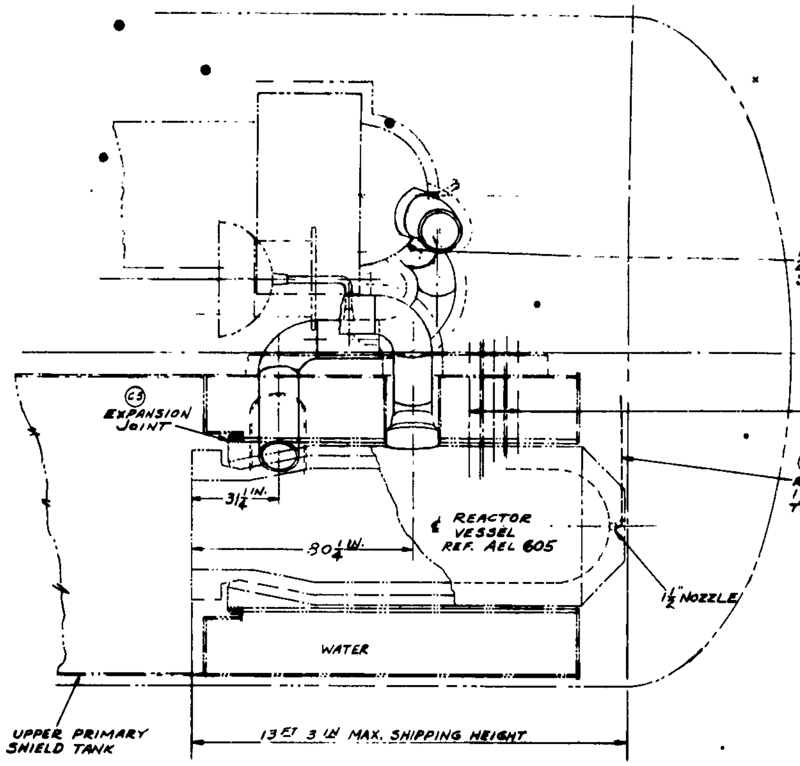


1" DRAIN LINE-USED AS PRIMARY BLOWDOWN.
22 IN
38 IN
B
SKID
STEAM GENERATOR
REFER TO DRGS. M11584-38 (OUTLINE) M11594-41 (DETAILS)
PUMP
20 FT. 0 IN INS. DIA. V.C.
PRESSURIZER REF M11594-40
CONTROL ROD DRIVES
STREAMING SHIELDING FOR REACTOR NOZZLES. REFER TO DRAWING M11594-94
36 IN
A
A
90° S.R. EL. 16"
62 IN
32½ IN CONCRETE @140#/FT³ MINIMUM NOT INCLUDING REINFORCEMENT.
9 FT 0½" O.D. SHIELD TANK.
REACTOR BLOWDOWN & DRAIN 1½" REDUCED TO 1"
½ IN EXPANSION JOINT MATERIAL.
B
1
TE9
1" COOL DOWN LINE.
REFER TO PUMP OUTLINE DRG. M11584-37
REACTOR TOP FLANGE SEE SECTION "B-B"
LEVEL INDICATOR CONTROLLER
TO INSTR. JUNCT. BOX
PRESSURIZER
3 IN PRESSURIZER CONNECTING LINE (PITCHED BACK TO MAIN)
31¼ IN.
76½ IN
90° L.R. EL 14"
45° L.R. EL. (14")
45° L.R. EL. (16")
90° L.R. EL (16)
9 FT 6 IN.
13 FT 3 IN REF.
BOTTOM OF SKID
ELBOWS WELDED DIRECTLY TO NOZZLES IN SHOP (WITHOUT SPUD EXTENSIONS)
SECTION "A-A"

A	REVISED & REDRAWN	
B	1½" DRAIN LINE ... WAS 1½" BOTTOM BLOWDOWN. STEAM GENERATOR REF. WAS ... DWG 47237, PRESSURIZER REF. WAS AES 313. ADDED 2" COOL DOWN LINE. PUMP REF. WAS WESTINGHOUSE DRG ... ADDED DELINEATION OF PRESSURIZER. REACTOR VESSEL REF. WAS AEL 603	
C	... ADDED VALVE ... REACTOR BLOWDOWN WAS 1½ ... ADDED 1½" NOZZLE ... EXPANSION JOINT ... TITLE WAS "VAPOR CONTAINER" ... ADDED NOTES "BOTTOM OF SKID" ... REACTOR TOP FLANGE ... ADDED TEMP. ...	
D	ADDED ... M11594-57 ...	

PROVIDE 1" BLOWDOWN & 2 INSPECTION PORTS IN STEAM GENERATOR CHANNEL.

SECTION "A-A"



SECTION "B-B"

2

LEGEND:-
T.E. = PRIMARY TEMPERATURE ELEMENT.
F.E. = PRIMARY FLOW ELEMENT.
F.X. = PRIMARY FLOW BLIND TRANSMITTER.

FOR PRIMARY SYSTEM EQUIPMENT ARRANGEMENT REFER TO DRG M11594-58

M11594-57 D

LIST OF MATERIAL

APPLICATION

QTY REQD

R. D. B.

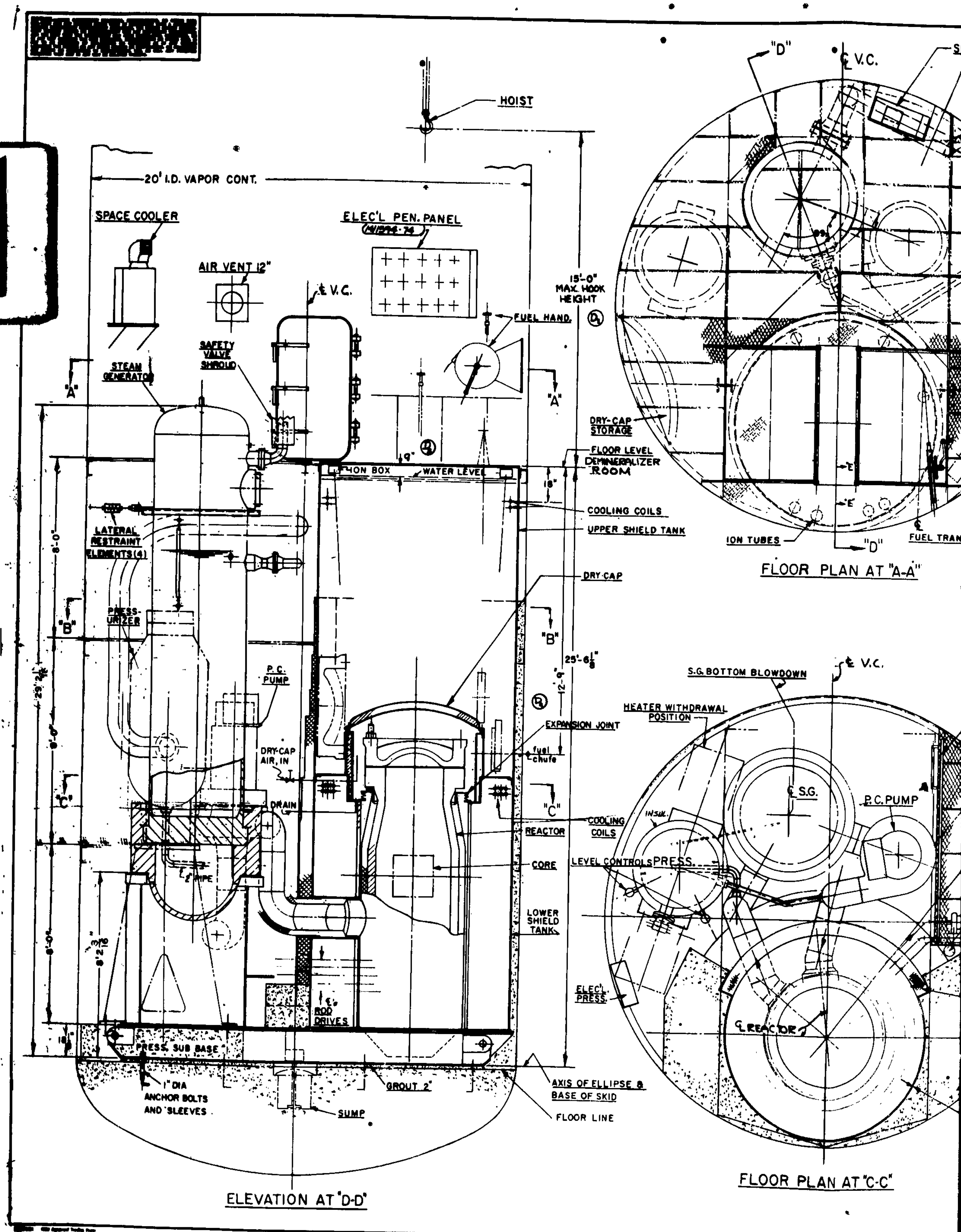
PRIMARY SYSTEM
PRIMARY PIPING
LAYOUT

U. S. ARMY ... CORPS OF ENGINEERS FORT BELVOIR, VA.

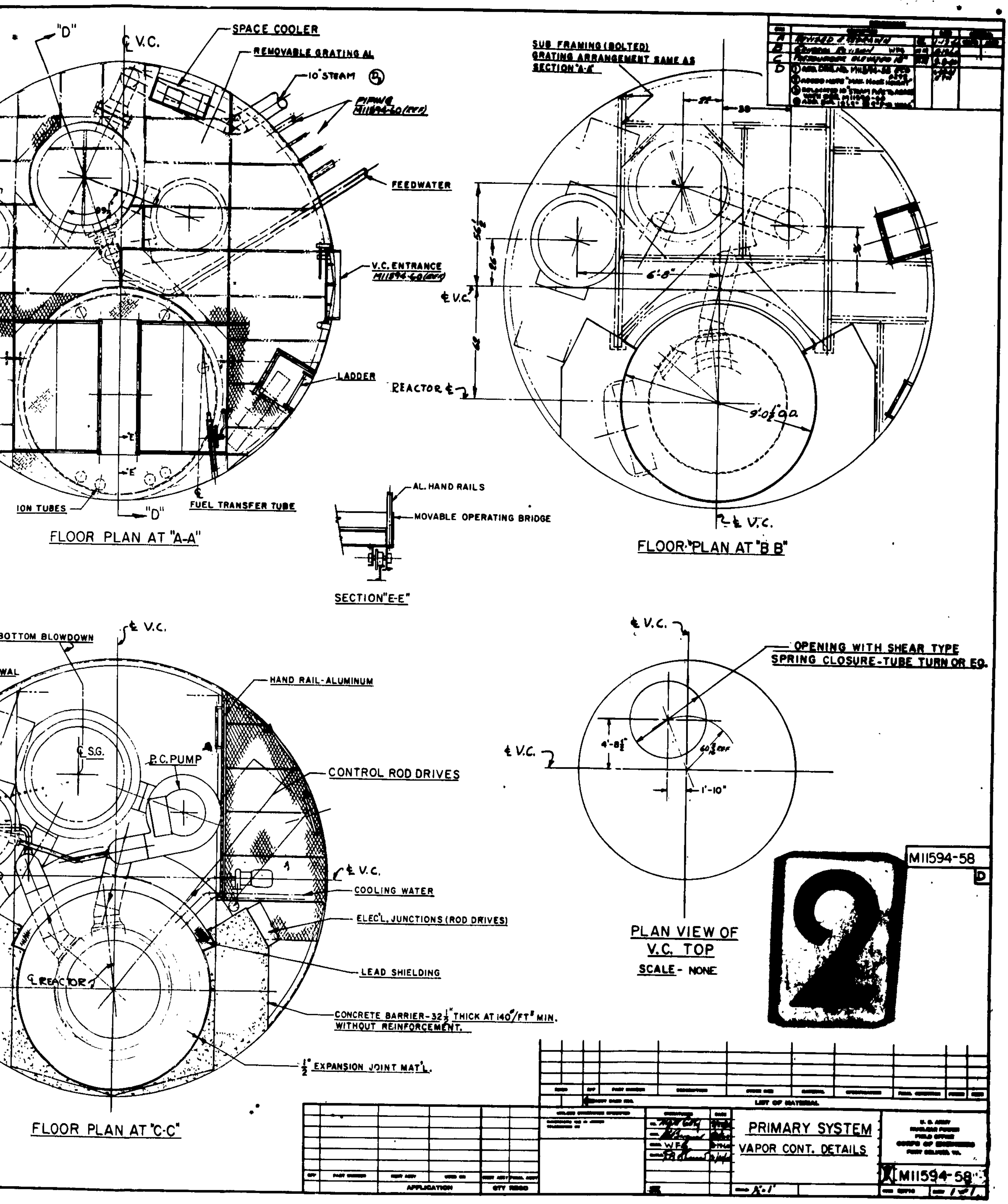
M11594-57

¼" = 1'-0"

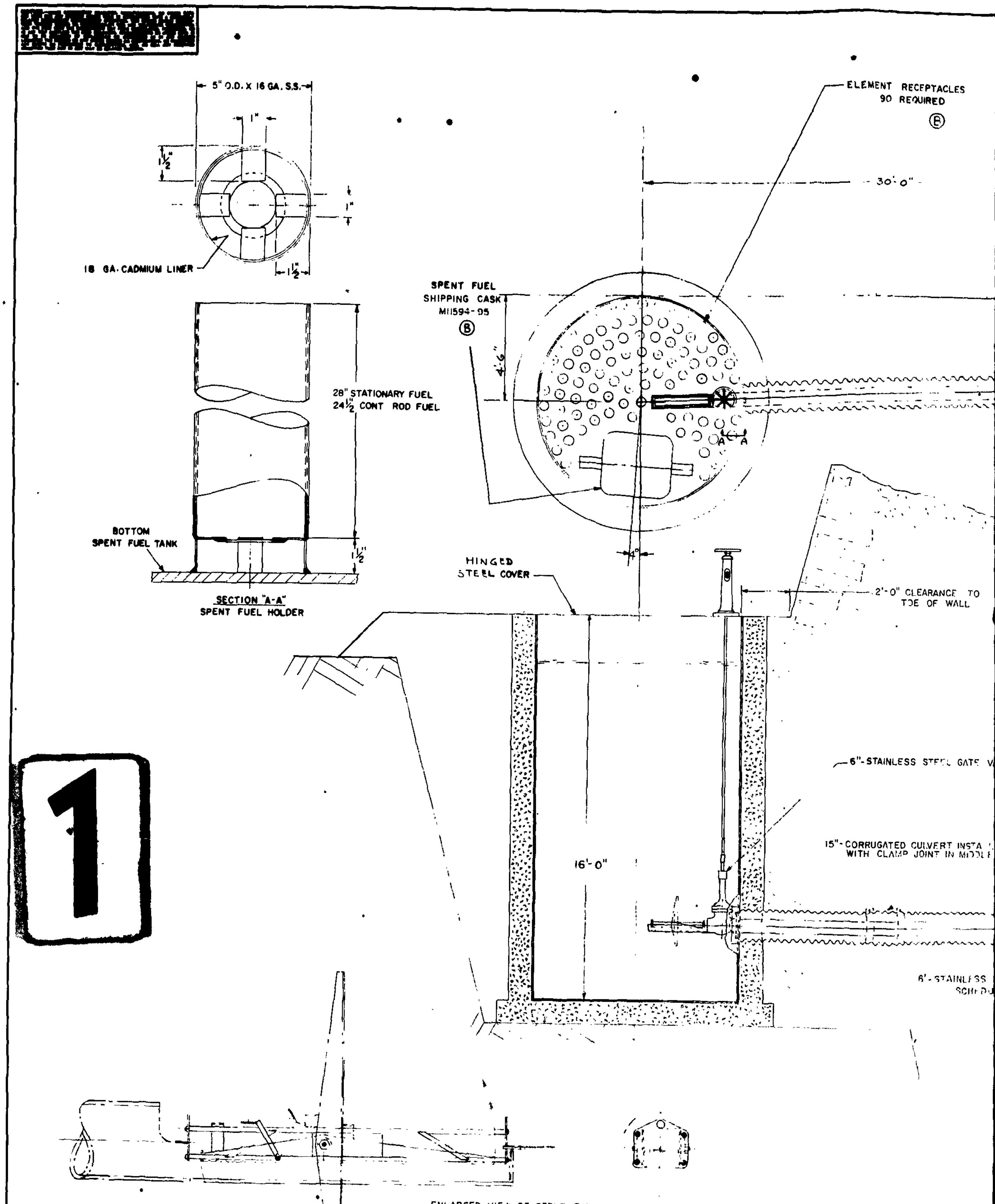
1



HOIST
20' I.D. VAPOR CONT.
SPACE COOLER
ELEC'L PEN. PANEL
AIR VENT 12"
℄ V.C.
15'-0" MAX. HOOK HEIGHT
FUEL HAND.
SAFETY VALVE SHROUD
STEAM GENERATOR
"A"
"A"
DRY-CAP STORAGE
FLOOR LEVEL DEMINERALIZER ROOM
ION BOX
WATER LEVEL
COOLING COILS
UPPER SHIELD TANK
LATERAL RESTRAINT ELEMENTS (4)
DRY-CAP
"B"
"B"
PRESS-URIZER
P.C. PUMP
25'-6 1/8"
EXPANSION JOINT
fuel chute
DRY-CAP AIR IN
DRAIN
"C"
"C"
COOLING COILS
REACTOR
CORE
LOWER SHIELD TANK
ROD DRIVES
PRESS. SUB BASE
1" DIA ANCHOR BOLTS AND SLEEVES
GROUT 2"
SUMP
AXIS OF ELLIPSE & BASE OF SKID
FLOOR LINE
8'-0"
8'-0"
8'-0"
18"
ELEVATION AT "D-D"
"D"
℄ V.C.
ION TUBES
"D"
FLOOR PLAN AT "A-A"
S.G. BOTTOM BLOWDOWN
℄ V.C.
HEATER WITHDRAWAL POSITION
℄ S.G.
P.C. PUMP
LEVEL CONTROLS
PRESS.
ELEC'L PRESS.
℄ REACTOR
FLOOR PLAN AT "C-C"

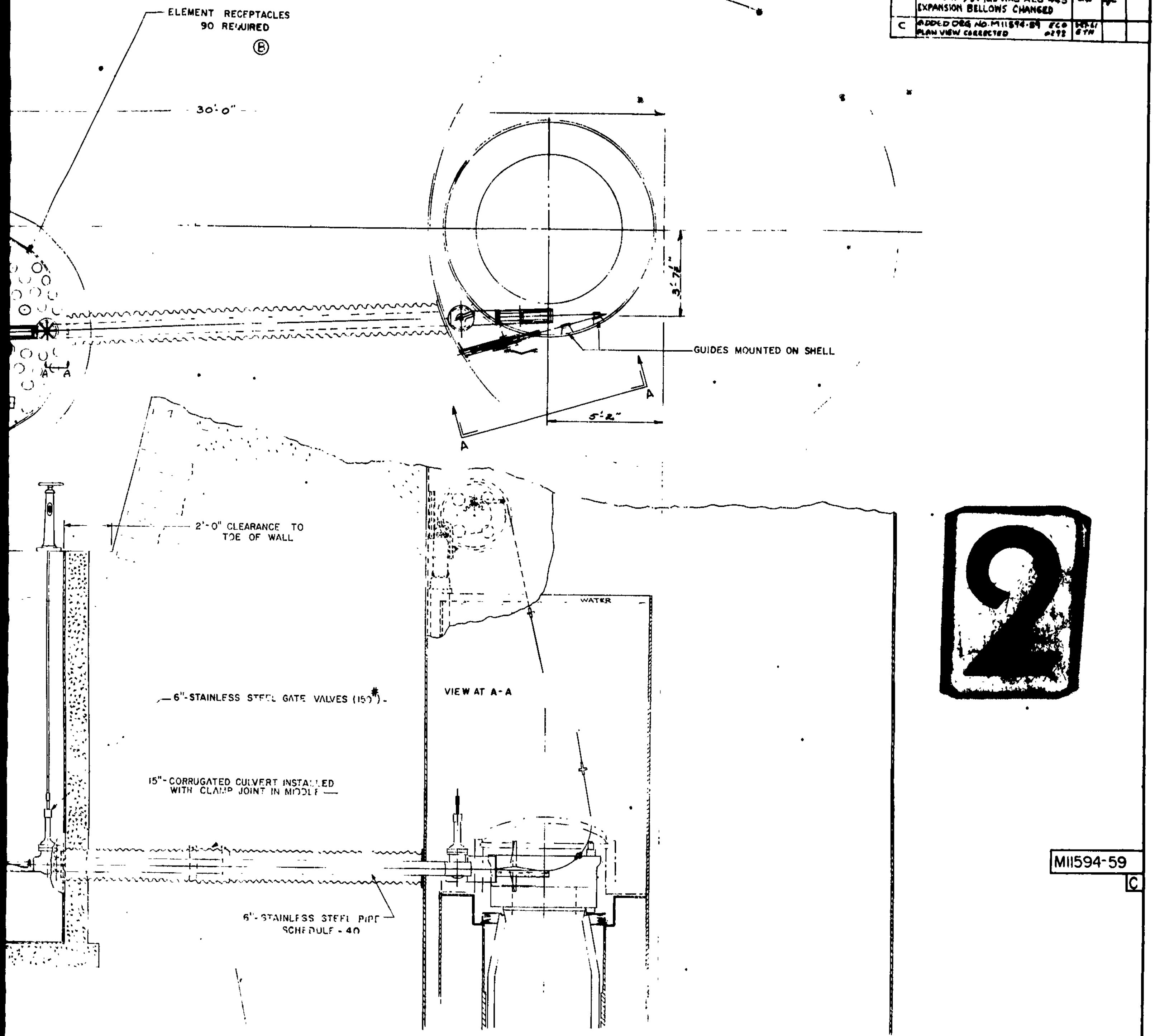


SPACE COOLER
REMOVABLE GRATING AL.
10" STEAM
PIPING M11594-60 (REF)
FEEDWATER
V.C. ENTRANCE M11594-60 (REF)
LADDER
FUEL TRANSFER TUBE
ION TUBES
"D"
℄ V.C.
FLOOR PLAN AT "A-A"
AL. HAND RAILS
MOVABLE OPERATING BRIDGE
SECTION "E-E"
SUB FRAMING (BOLTED) GRATING ARRANGEMENT SAME AS SECTION "A-A"
REACTOR ℄
6'-8"
9'-0 1/2" O.D.
FLOOR PLAN AT "B B"
BOTTOM BLOWDOWN
HAND RAIL-ALUMINUM
℄ S.G.
P.C. PUMP
CONTROL ROD DRIVES
COOLING WATER
ELEC'L. JUNCTIONS (ROD DRIVES)
LEAD SHIELDING
℄ REACTOR
CONCRETE BARRIER-32 1/2" THICK AT 140#/FT³ MIN. WITHOUT REINFORCEMENT.
1/2" EXPANSION JOINT MAT'L.
FLOOR PLAN AT "C-C"
OPENING WITH SHEAR TYPE SPRING CLOSURE-TUBE TURN OR EQ.
4'-8 1/2"
1'-10"
PLAN VIEW OF V.C. TOP
SCALE - NONE
M11594-58
D
2
LIST OF MATERIAL
APPLICATION
PRIMARY SYSTEM
VAPOR CONT. DETAILS
M11594-58



ENLARGED VIEW OF SPENT FUEL
SLED & BASKET
SCALE 3" = 1 FOOT
REF DRG. MI1594
B

A	REVISED DUE TO NUMEROUS CHANGES	7-20-60		
B	APP DWG ... REMOVED F9-48-2053 WAS R9-48-2053 DWG NO. F9-34 1013 WAS AES-445 EXPANSION BELLOWS CHANGED			
C	ADDED DRG NO. M11594-59 PLAN VIEW CORRECTED ECO #292			



M11594-59
C

LIST OF MATERIAL

APPLICATION | QTY REQD

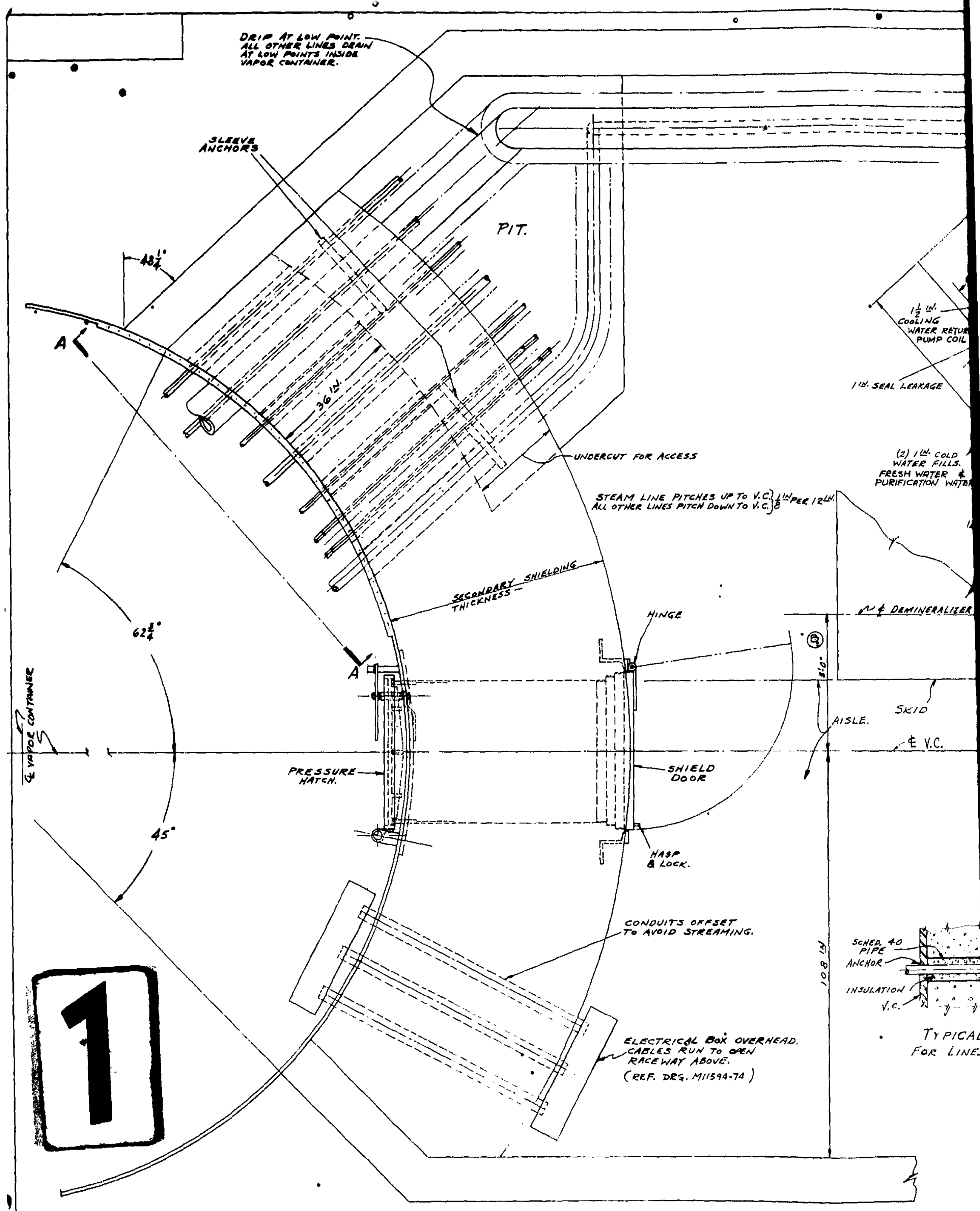
Williams 7-20-60

PRIMARY SYSTEM
FUEL TRANSFER EQUIPMENT

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

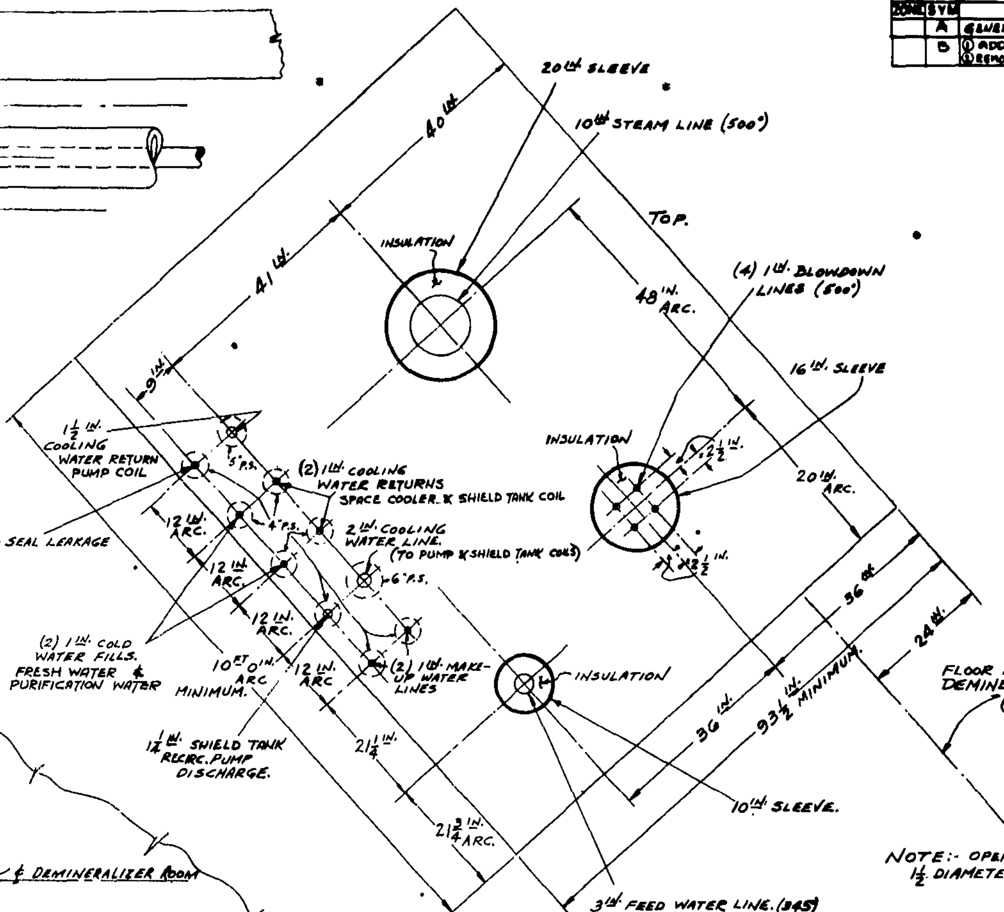
M11594-59

SCALE ¾" = 1 FT.


DRIP AT LOW POINT. ALL OTHER LINES DRAIN AT LOW POINTS INSIDE VAPOR CONTAINER.
SLEEVE ANCHORS
PIT.
$48\frac{1}{4}°$
A
36 IN.
$1\frac{1}{2}$ IN. COOLING WATER RETUR PUMP COIL
1 IN. SEAL LEAKAGE
UNDERCUT FOR ACCESS
(2) 1 IN. COLD WATER FILLS. FRESH WATER & PURIFICATION WATE
STEAM LINE PITCHES UP TO V.C. ALL OTHER LINES PITCH DOWN TO V.C. $\frac{1}{8}$ IN. PER 12 IN.
SECONDARY SHIELDING THICKNESS
$62\frac{3}{4}°$
HINGE
℄ DEMINERALIZER
8'-0"
A
SKID
AISLE.
℄ VAPOR CONTAINER
℄ V.C.
PRESSURE HATCH.
SHIELD DOOR
45°
HASP & LOCK.
CONDUITS OFFSET TO AVOID STREAMING.
SCHED. 40 PIPE
ANCHOR
108 IN.
INSULATION
V.C.
TYPICAL
FOR LINES
ELECTRICAL BOX OVERHEAD. CABLES RUN TO OPEN RACEWAY ABOVE.
(REF. DRG. M11594-74)
1

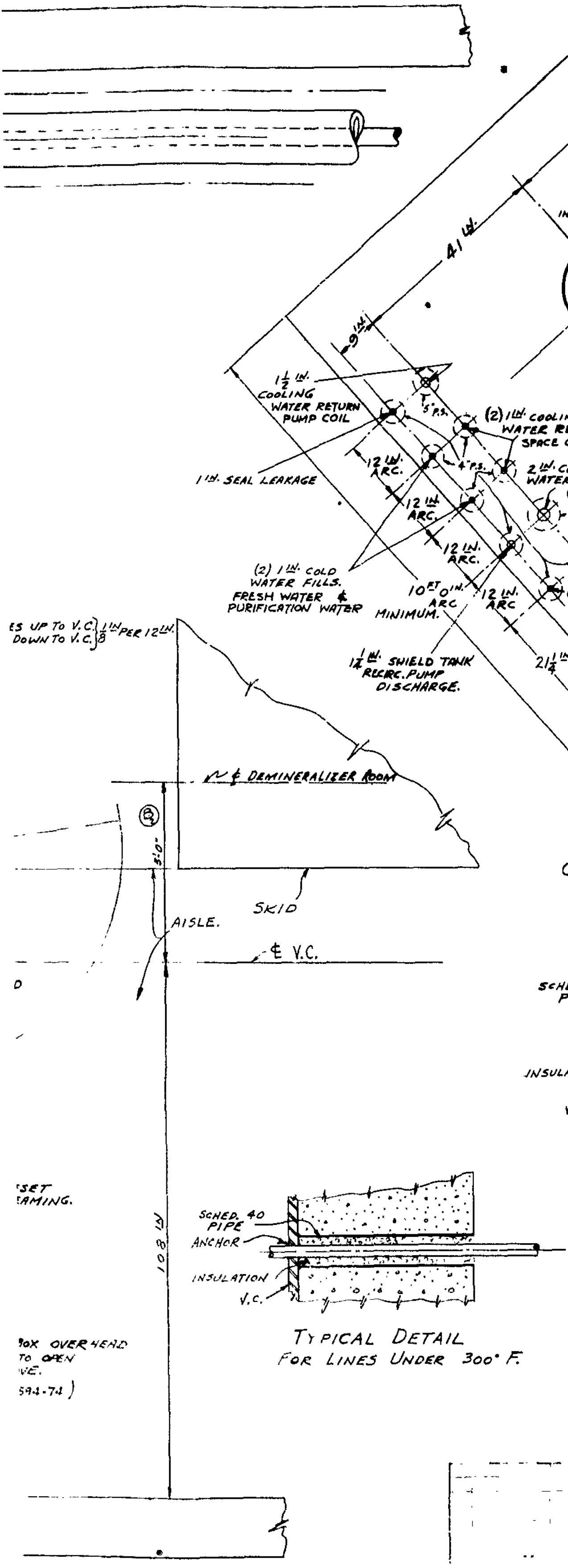
REVISIONS

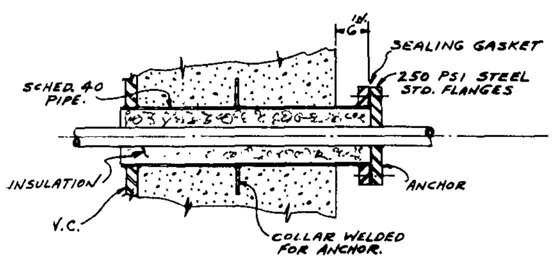
ZONE	SYM	DESCRIPTION	DATE	APPROVAL
	A	GENERAL REVISION	9-4-60	
	B	① ADD. DWG. NO M11594-60 ECO ② REMOVE 12" DIM. 0272	3-27-61	



NOTE:- OPENING SPACINGS ARE 1½ DIAMETERS OF CUTOUT.

VIEW "A-A"
CODE SPACING



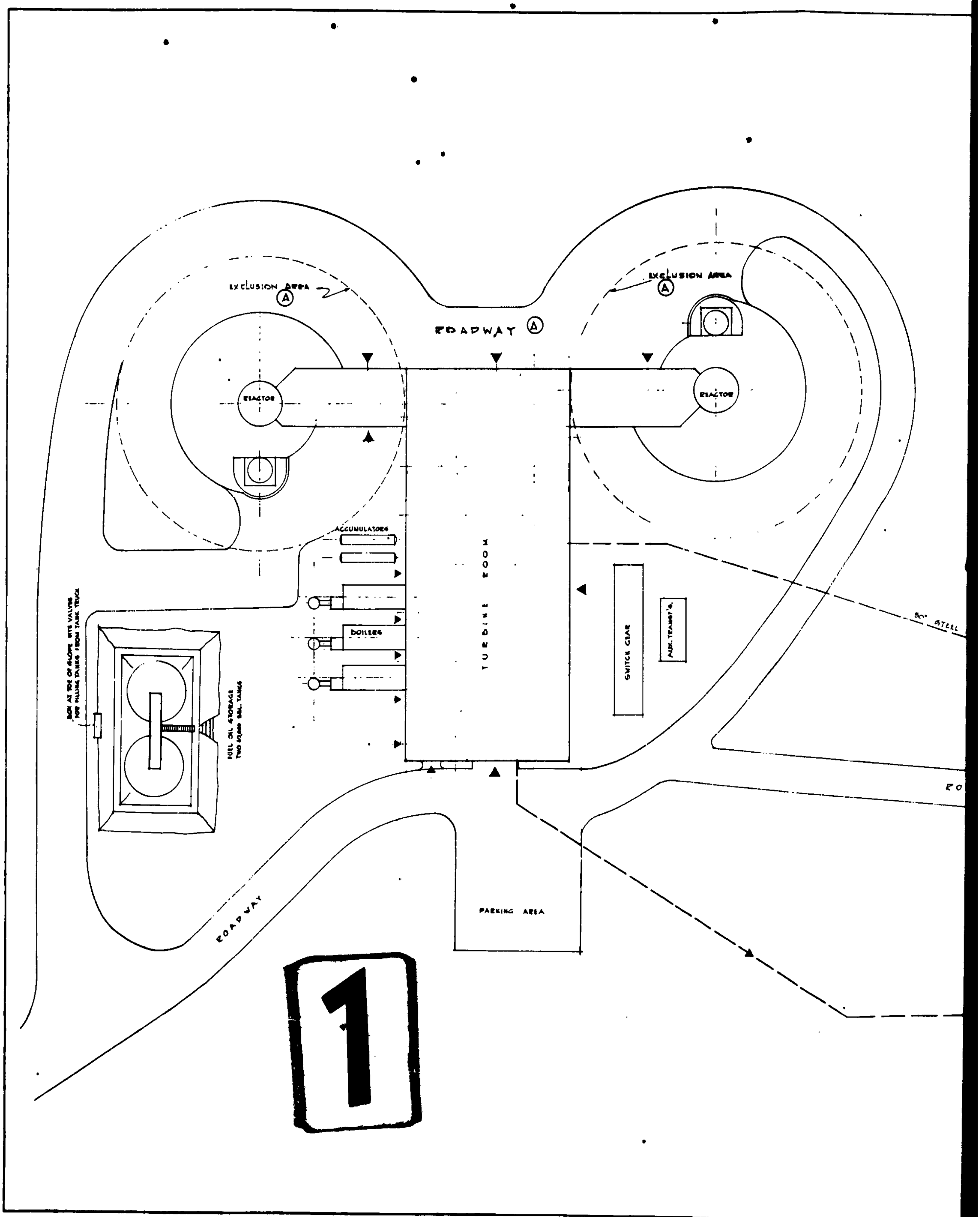


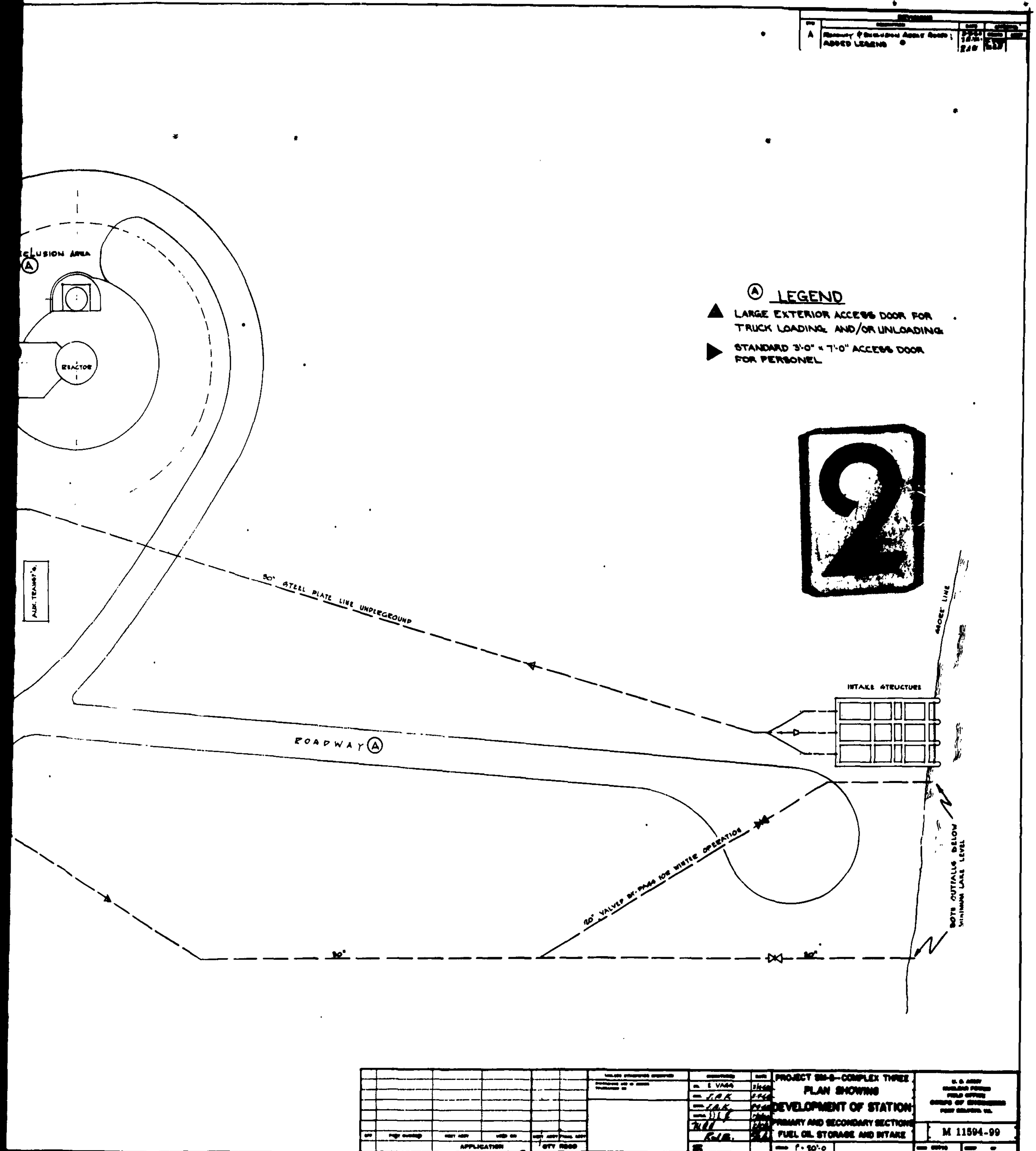
TYPICAL DETAIL
FOR LINES OVER 300° F.

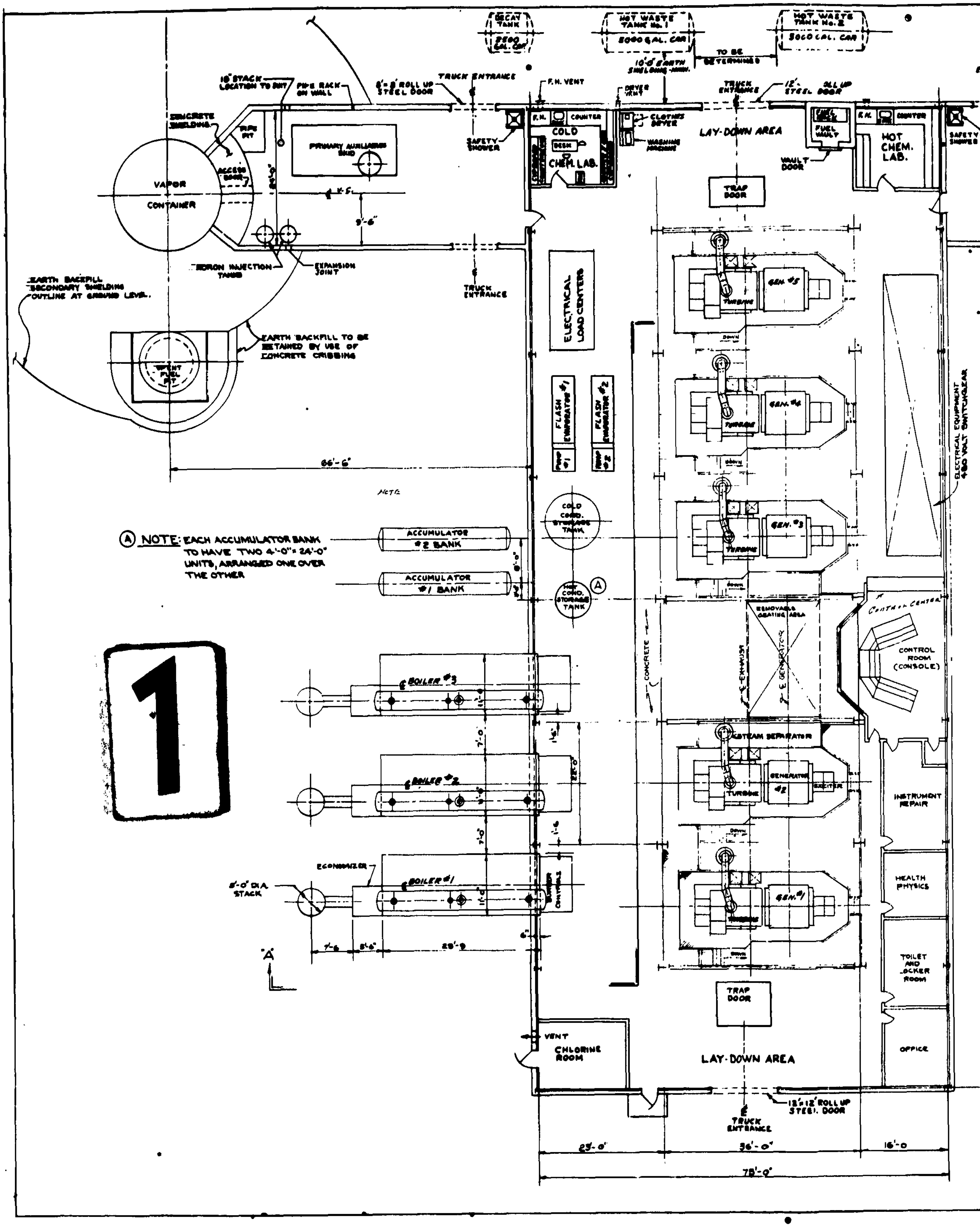
TYPICAL DETAIL
FOR LINES UNDER 300° F.

2

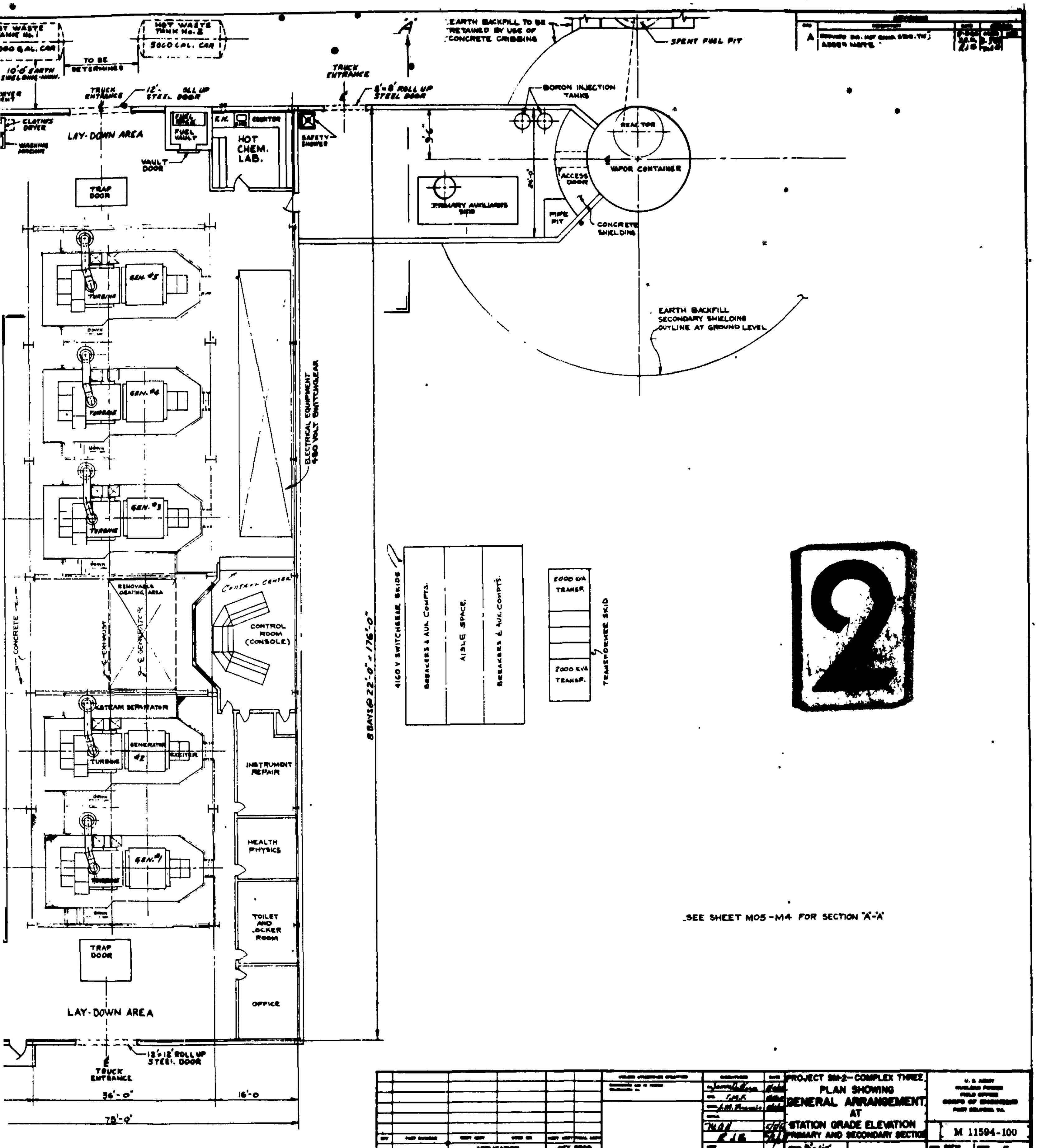
R.D.B.	PRIMARY SYSTEM VAPOR CONTAINER ENTRIES
D.D.F.	U. S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR VA
	M11594-60

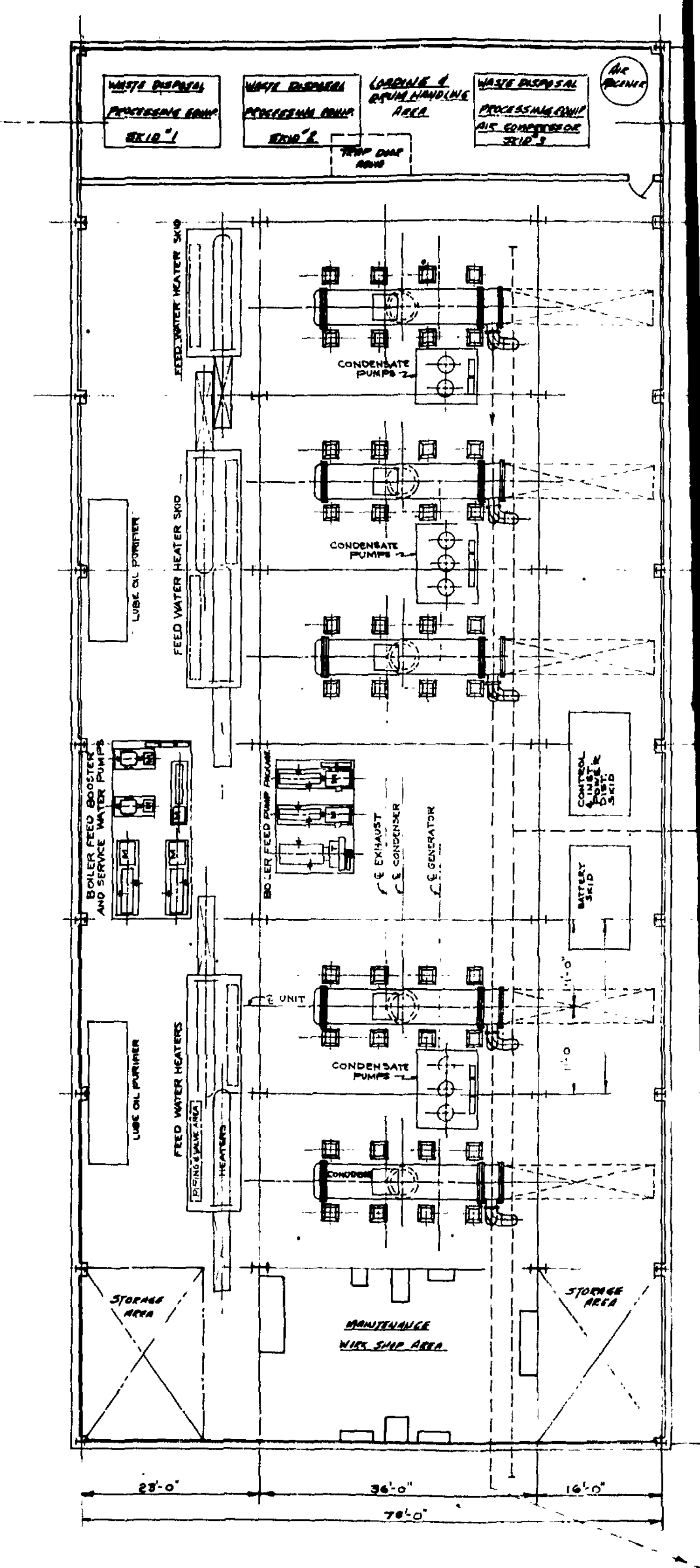

EXCLUSION AREA A
ROADWAY A
EXCLUSION AREA A
REACTOR
REACTOR
ACCUMULATORS
BOILERS
TURBINE ROOM
SWITCH GEAR
AUX. TRANSF'S.
BOX AT TOP OF SLOPE WITH VALVES FOR FILLING TANKS FROM TANK TRUCK
FUEL OIL STORAGE
PARKING AREA
ROADWAY
1


A
ADDED LEGEND
EXCLUSION AREA (A)
REACTOR
AUX. TRANS'S.
(A) LEGEND
LARGE EXTERIOR ACCESS DOOR FOR TRUCK LOADING AND/OR UNLOADING
STANDARD 3'-0" × 7'-0" ACCESS DOOR FOR PERSONEL
2
30" STEEL PLATE LINE UNDERGROUND
SHORE LINE
INTAKE STRUCTURE
ROADWAY (A)
30" VALVED BY-PASS FOR WINTER OPERATION
BOTH OUTFALLS BELOW MINIMUM LAKE LEVEL
30"
30"
PROJECT SM-1—COMPLEX THREE
PLAN SHOWING
DEVELOPMENT OF STATION
PRIMARY AND SECONDARY SECTIONS
FUEL OIL STORAGE AND INTAKE
APPLICATION
1" = 30'-0
M 11594-99



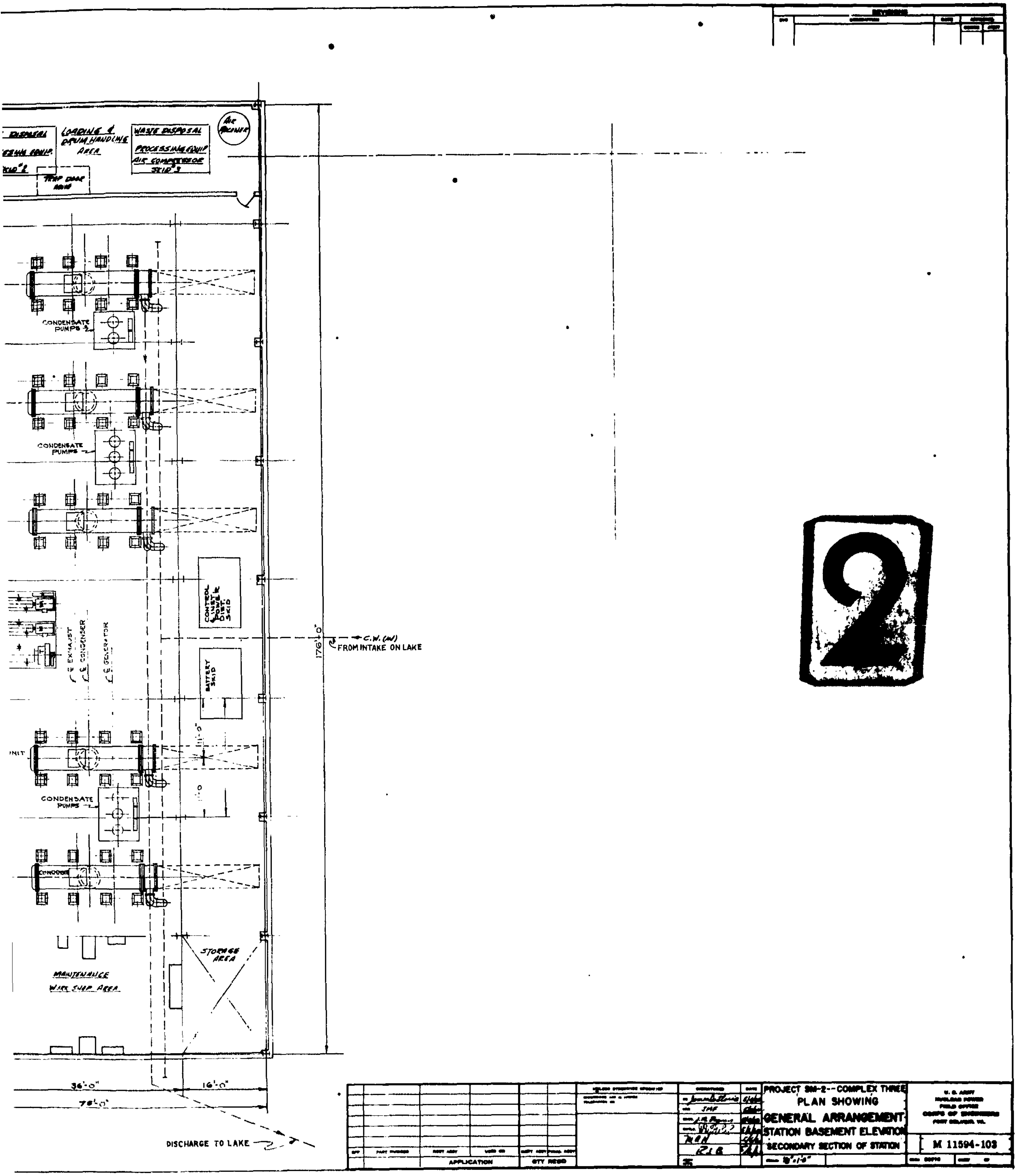
DECAY TANK
2500 GAL. CAP.
HOT WASTE TANK No. 1
5000 GAL. CAP.
HOT WASTE TANK No. 2
5000 GAL. CAP.
10'-0" EARTH SHIELDING
TO BE DETERMINED
18" STACK LOCATION TO SUIT
PIPE RACK ON WALL
8'×8' ROLL UP STEEL DOOR
TRUCK ENTRANCE
F.H. VENT
DRYER VENT
TRUCK ENTRANCE
12' ROLL UP STEEL DOOR
CONCRETE SHIELDING
PIPE PIT
ACCESS DOOR
PRIMARY AUXILIARIES SKID
VAPOR CONTAINER
SAFETY SHOWER
F.H.
COUNTER
COLD CHEM. LAB.
DESK
CLOTHES DRYER
WASHING MACHINE
LAY-DOWN AREA
FUEL VAULT
VAULT DOOR
HOT CHEM. LAB.
SAFETY SHOWER
TRAP DOOR
9'-6"
BORON INJECTION TANKS
EXPANSION JOINT
TRUCK ENTRANCE
EARTH BACKFILL SECONDARY SHIELDING OUTLINE AT GROUND LEVEL.
EARTH BACKFILL TO BE RETAINED BY USE OF CONCRETE CRIBBING
SPENT FUEL PIT
ELECTRICAL LOAD CENTERS
GEN. #5
TURBINE
GEN. #4
GEN. #3
FLASH EVAPORATOR #1
FLASH EVAPORATOR #2
PUMP #1
PUMP #2
66'-6"
COLD COND. STORAGE TANK
HOT COND. STORAGE TANK
ELECTRICAL EQUIPMENT 480 VOLT SWITCHGEAR
Ⓐ NOTE: EACH ACCUMULATOR BANK TO HAVE TWO 4'-0"×24'-0" UNITS, ARRANGED ONE OVER THE OTHER
ACCUMULATOR #2 BANK
ACCUMULATOR #1 BANK
REMOVABLE GRATING AREA
CONTROL CENTER
CONTROL ROOM (CONSOLE)
1
CONCRETE
BOILER #3
BOILER #2
BOILER #1
STEAM SEPARATOR
GENERATOR #2
EXCITER
INSTRUMENT REPAIR
ECONOMIZER
5'-0" DIA. STACK
GEN. #1
HEALTH PHYSICS
TOILET AND LOCKER ROOM
"A"
TRAP DOOR
VENT
CHLORINE ROOM
LAY-DOWN AREA
OFFICE
12'×12' ROLL UP STEEL DOOR
TRUCK ENTRANCE
23'-0"
36'-0"
16'-0
78'-0"


HOT WASTE TANK No. 2
5000 GAL. CAP.
TO BE DETERMINED
10'-0" EARTH SHIELDING
TRUCK ENTRANCE
12' STEEL DOOR
A
EARTH BACKFILL TO BE RETAINED BY USE OF CONCRETE CRIBBING
SPENT FUEL PIT
TRUCK ENTRANCE
8'x8' ROLL UP STEEL DOOR
BORON INJECTION TANKS
REACTOR
VAPOR CONTAINER
CLOTHES DRYER
WASHING MACHINE
LAY-DOWN AREA
FUEL VAULT
VAULT DOOR
COUNTER
HOT CHEM. LAB.
SAFETY SHOWER
ACCESS DOOR
PRIMARY AUXILIARIES SKID
PIPE PIT
CONCRETE SHIELDING
TRAP DOOR
GEN. #5
TURBINE
EARTH BACKFILL SECONDARY SHIELDING OUTLINE AT GROUND LEVEL
GEN. #4
ELECTRICAL EQUIPMENT 480 VOLT SWITCHGEAR
GEN. #3
REMOVABLE GRATING AREA
CONTROL CENTER
CONTROL ROOM (CONSOLE)
CONCRETE
8 BAYS @ 22'-0" = 176'-0"
4160 V SWITCHGEAR SKIDS
BREAKERS & AUX. COMPTS.
AISLE SPACE
BREAKERS & AUX. COMPTS.
2000 KVA TRANSF.
2000 KVA TRANSF.
TRANSFORMER SKID
STEAM SEPARATOR
GENERATOR #2
EXCITER
TURBINE
INSTRUMENT REPAIR
GEN. #1
HEALTH PHYSICS
TOILET AND LOCKER ROOM
OFFICE
TRAP DOOR
LAY-DOWN AREA
12'x12' ROLL UP STEEL DOOR
TRUCK ENTRANCE
36'-0"
16'-0
75'-9"
SEE SHEET M05-M4 FOR SECTION "A-A"
2
PROJECT SM-2—COMPLEX THREE
PLAN SHOWING
GENERAL ARRANGEMENT
AT
STATION GRADE ELEVATION
PRIMARY AND SECONDARY SECTION
APPLICATION
M 11594-100



WASTE DISPOSAL
PROCESSING EQUIP
SKID #1
WASTE DISPOSAL
PROCESSING EQUIP
SKID #2
LOADING & DRUM HANDLING AREA
WASTE DISPOSAL
PROCESSING EQUIP
AIR COMPRESSOR
SKID 3
FEED WATER HEATER SKID
CONDENSATE PUMPS
LUBE OIL PURIFIER
FEED WATER HEATER SKID
BOILER FEED BOOSTER AND SERVICE WATER PUMPS
BOILER FEED PUMP PACKAGE
℄ EXHAUST
℄ CONDENSER
℄ GENERATOR
CONTROL
BATTERY SKID
℄ UNIT
FEED WATER HEATERS
HEATERS
STORAGE AREA
MAINTENANCE
WORK SHOP AREA
STORAGE AREA
28'-0"
36'-0"
16'-0"
78'-0"
DISCHARGE TO LAKE

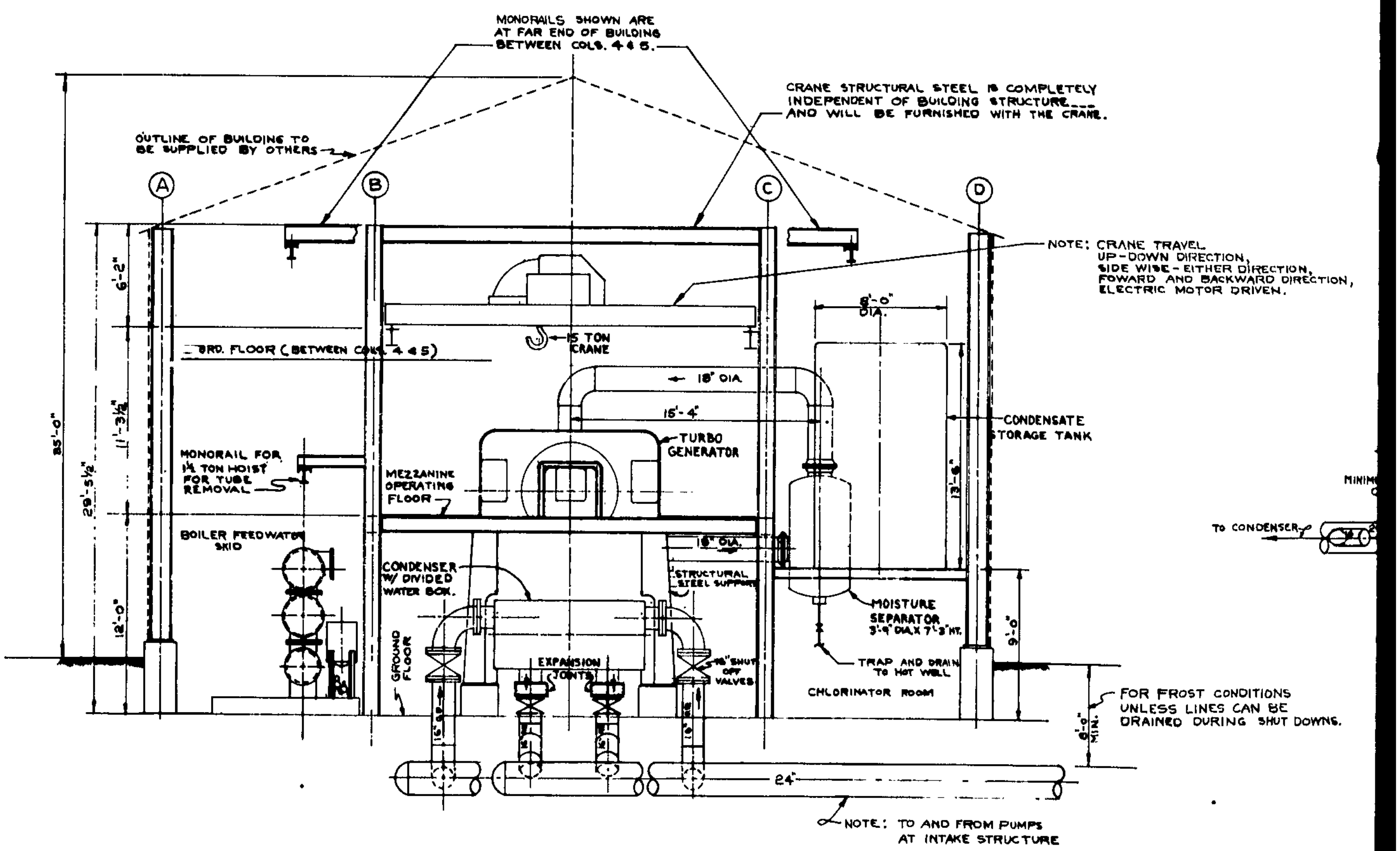
1



LOADING & DRUM HANDLING AREA
WASTE DISPOSAL
PROCESSING EQUIP.
AIR COMPRESSOR SKID 3
AIR RECEIVER
CONDENSATE PUMPS
CONDENSATE PUMPS
CONTROL LINES & DIST. SKID
BATTERY SKID
℄ EXHAUST
℄ CONDENSER
℄ GENERATOR
CONDENSATE PUMPS
176'-0"
C.W. (IN) FROM INTAKE ON LAKE
STORAGE AREA
MAINTENANCE WORK SHOP AREA
36'-0"
16'-0"
76'-0"
DISCHARGE TO LAKE
APPLICATION
QTY REQD
PROJECT SM-2--COMPLEX THREE
PLAN SHOWING
GENERAL ARRANGEMENT
STATION BASEMENT ELEVATION
SECONDARY SECTION OF STATION
U. S. ARMY
M 11594-103

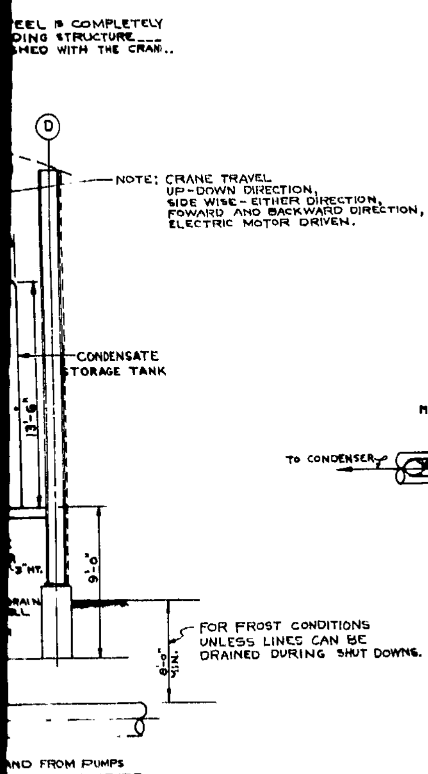
2

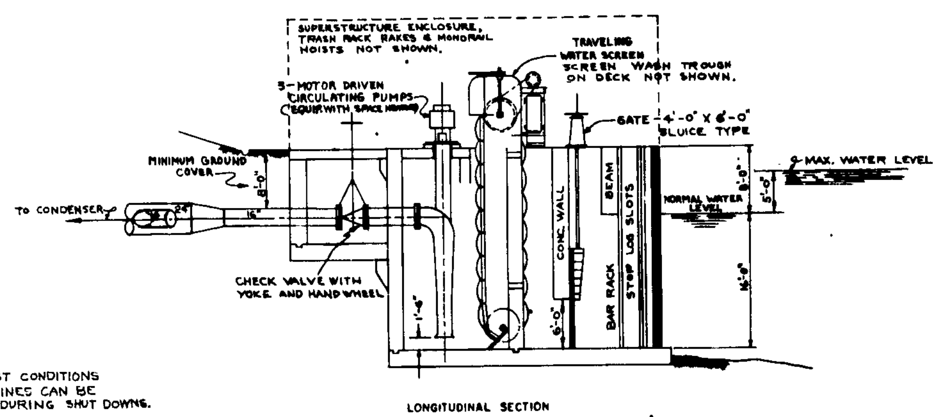
1



SECTION A-A
AT COL. LINE 1a
SCALE 1/4"=1'-0"
FOR LOCATION OF THIS SECTION SEE M03-M2

2





LONGITUDINAL SECTION
CIRCULATING WATER
INTAKE STRUCTURE
SUPERSTRUCTURE ENCLOSURE
TRASH RACK RAKE AND MONORAIL HOISTS
NOT SHOWN
SCALE: 1/8" = 1'-0"

NOTE:

DESIGN OF TURBO-GENERATOR CONDENSER IS BASED ON UTILIZATION OF A CONDENSER STRUCTURALLY DESIGNED TO SUPPORT A TURBO-GENERATOR, OR SEPERATE SUPPORT IS PROVIDED BY TURBINE MANUFACTURER AND CONDENSER IS HUNG FROM TURBINE EXHAUST.

MAX. ALLOWABLE ELEVATION OF ANY STRUCTURE CAN NOT EXCEED 35'-0" ABOVE GRADE.

SCREEN HOUSE LOCATED ABOUT 500 FEET FROM SECONDARY SYSTEM BUILDING.

MOTOR DRIVEN CIRCULATING COOLING WATER PUMPS WILL BE BASED ON 3-50% UNITS INSTALLED, 2 OPERATING AND ONE STANDBY.

SM2--ONE 7,500 KW UNIT
TYPICAL SECTIONS
SHOWING
POWER PLANT & INTAKE STRUCTURE
SECONDARY EQUIPMENT INSTALLATION
INTAKE RACKS SCREENS AND PUMPS

SCALE AS SHOWN

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M 11594-8

3.0 HEAT BALANCE

Heat losses by conduction, convection, radiation, and blowdown from the primary system were calculated and these losses are:

	Btu/hr
Reactor heat loss	65360
Steam generator heat loss	14220
Pressurizer heat loss	7100
Primary piping heat loss	4980
Primary blowdown loss	235000
Total heat loss	326660

The reactor heat loss includes 50, 690 Btu/hr heat loss due to gamma radiation to the primary shielding.

The heat input to the primary coolant due to primary pump work is 661, 000 Btu/hr, so that the net heat from losses and pump work is a gain of 334, 340 Btu/hr.

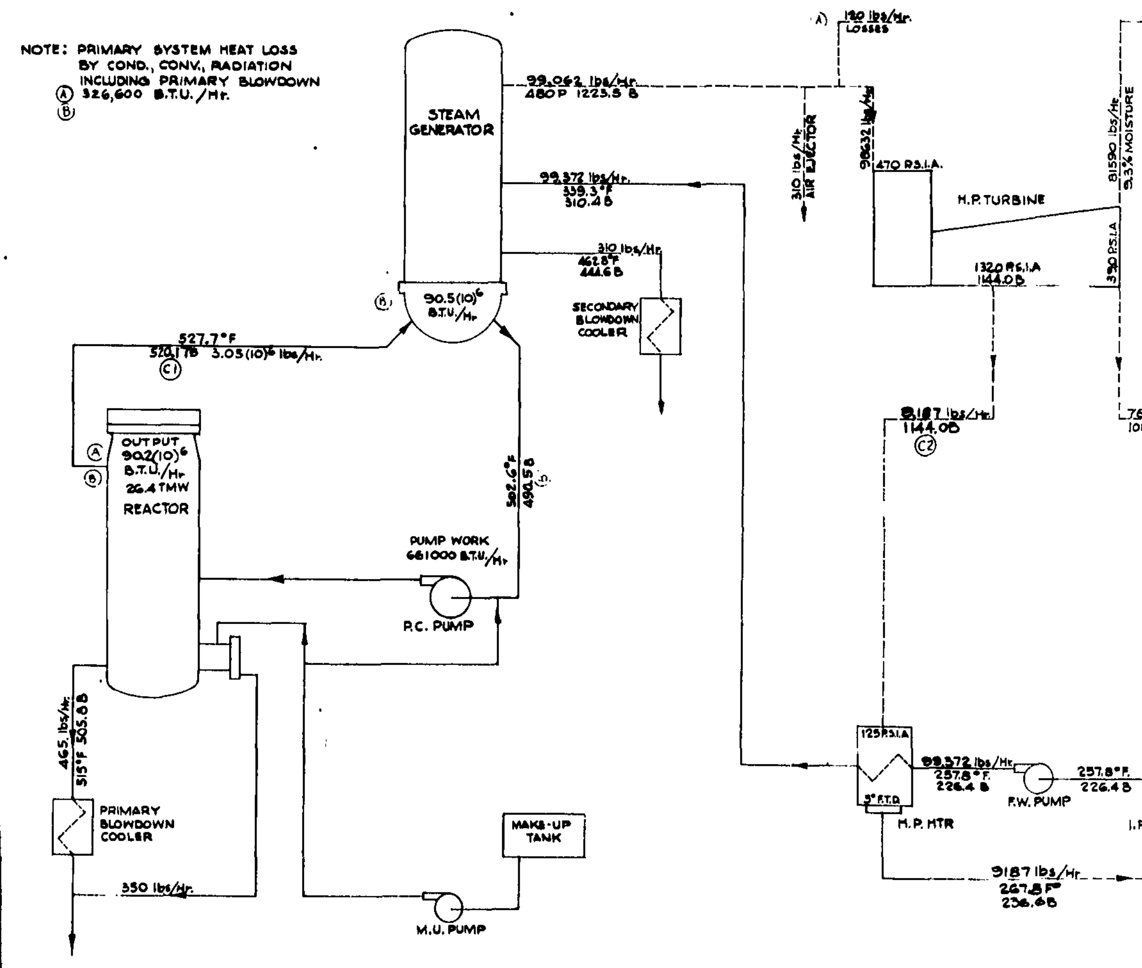
Heat balances have been prepared for the single reactor, single turbine-generator set for the condition of rated heat transferred at the steam generator. The heat output at the steam generator at rated conditions is 90. 5 x 10^6 Btu/hr and the gross electrical output is 7511 kw. The heat balance is shown on drawing M 11594-50. Heat balances have also been prepared to show cycle conditions for the system of two reactors supplying three turbines for a net output of 12, 000 kw and four turbines for a net output of 12, 000 kw. These heat balances are on drawings M 11594-101 and M 11594-102.

The following is a summary of the primary heat balances:

	Single Reactor Single T-G Set 7511 KW Gross Output	Two Reactor 3 T-G Sets 12000 KW Gross Output	Two Reactor 4 T-G Sets 12000 KW Gross Output
Steam Flow	99, 062 lb/hr	90, 545 lb/hr	91, 690 lb/hr
Feedwater Flow	99, 372 lb/hr	90, 855 lb/hr	92, 000 lb/hr
Steam Gen. Output	90. 5 $(10)^{-6}$ Btu/hr	84. 42 $(10)^6$ Btu/hr	87. 36 $(10)^6$ Btu/hr
Primary System Losses	326660 Btu/hr	326660 Btu/hr	326660 Btu/hr
Primary Pump Work	661000 Btu/hr	661000 Btu/hr	661000 Btu/hr
Reactor Output	90. 2 $(10)^6$ Btu/hr	84. 10 $(10)^6$ Btu/hr	87. 03 $(10)^6$ Btu/hr
Primary Coolant Flow	3. 05 $(10)^6$ lb/hr	3. 05 $(10)^6$ lb/hr	3. 05 $(10)^6$ lb/hr

Calculations for the primary heat balance are shown in Volume 2, Section 6. 0, of this report.

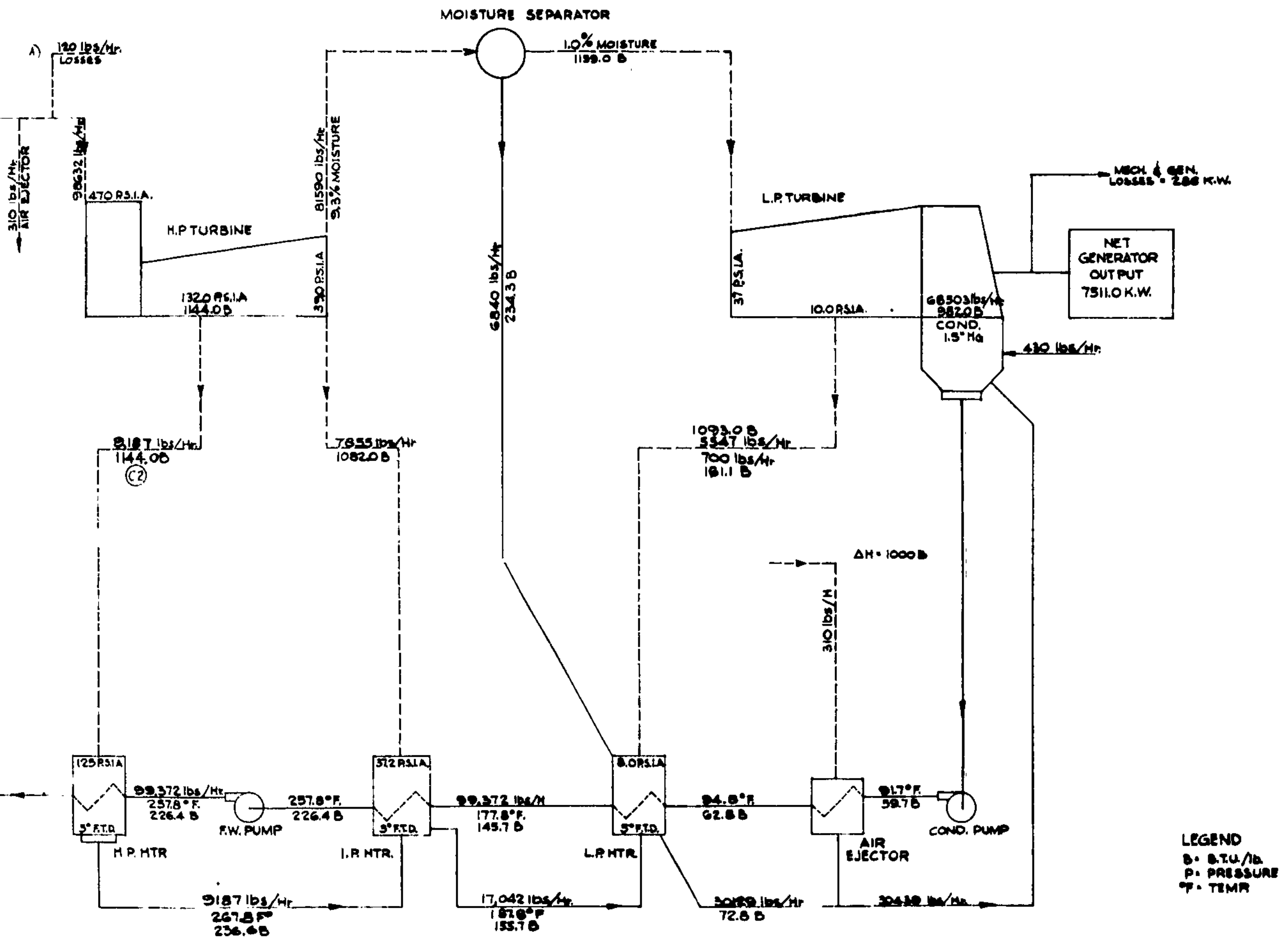
[redacted]



NOTE: PRIMARY SYSTEM HEAT LOSS
BY COND., CONV., RADIATION
INCLUDING PRIMARY BLOWDOWN
A 326,600 B.T.U./Hr.
B
STEAM GENERATOR
99,062 lbs/Hr.
480P 1223.5 B
99,372 lbs/Hr.
339.3°F
310.4 B
310 lbs/Hr.
462.8°F
444.6 B
SECONDARY BLOWDOWN COOLER
90.5(10)^6 B.T.U./Hr
527.7°F
520.17B 3.05(10)^6 lbs/Hr.
C1
OUTPUT
90.2(10)^6
B.T.U./Hr
26.4 TMW
REACTOR
A
B
502.6°F
490.5 B
PUMP WORK
661000 B.T.U./Hr
P.C. PUMP
465. lbs/Hr.
515°F 505.8 B
PRIMARY BLOWDOWN COOLER
350 lbs/Hr.
MAKE-UP TANK
M.U. PUMP
A
120 lbs/Hr.
LOSSES
310 lbs/Hr.
AIR EJECTOR
98632 lbs/Hr
470 P.S.I.A.
H.P. TURBINE
81590 lbs/Hr
9.3% MOISTURE
390 P.S.I.A
1320 P.S.I.A
1144.0 B
9187 lbs/Hr.
1144.0 B
C2
125 P.S.I.A
5° F.T.D.
H.P. HTR
99,372 lbs/Hr
257.8°F.
226.4 B
F.W. PUMP
257.8°F.
226.4 B
9187 lbs/Hr
267.8 F°
236.6 B

1

A	[illegible]	7-14-60		
B	AT REACTOR: 90.2 (10)[6] WAS 90.0(10)[6]; IN NOTE 326,600 [illegible] WAS 327 gpm; AT STM GEN. [illegible] WAS [illegible]; AT LINE TO AC. PUMP 302.6°F WAS 300.5°F & 490.5 B WAS 488.0B.	9-16-60 (T.D.)	JA	
C	① 520.178 WAS 520.278; ② 1144.0B WAS 1145.2B; ③ 6000eK.W. WAS 6600eK.W. (ECO 4122)	8-22-60	JA	



2

$$\text{HEAT RATE} = \frac{99{,}062\,(1223.5 - 310.4) + 310\,(444.6 - 310.4)}{7{,}511\ \text{eKW}} = 12{,}048\ \text{B.T.U./KWHR}$$

$$\text{EFFICIENCY} = \frac{3412.7}{12{,}048} = 28.33\%$$

LIST OF MATERIAL

APPLICATION

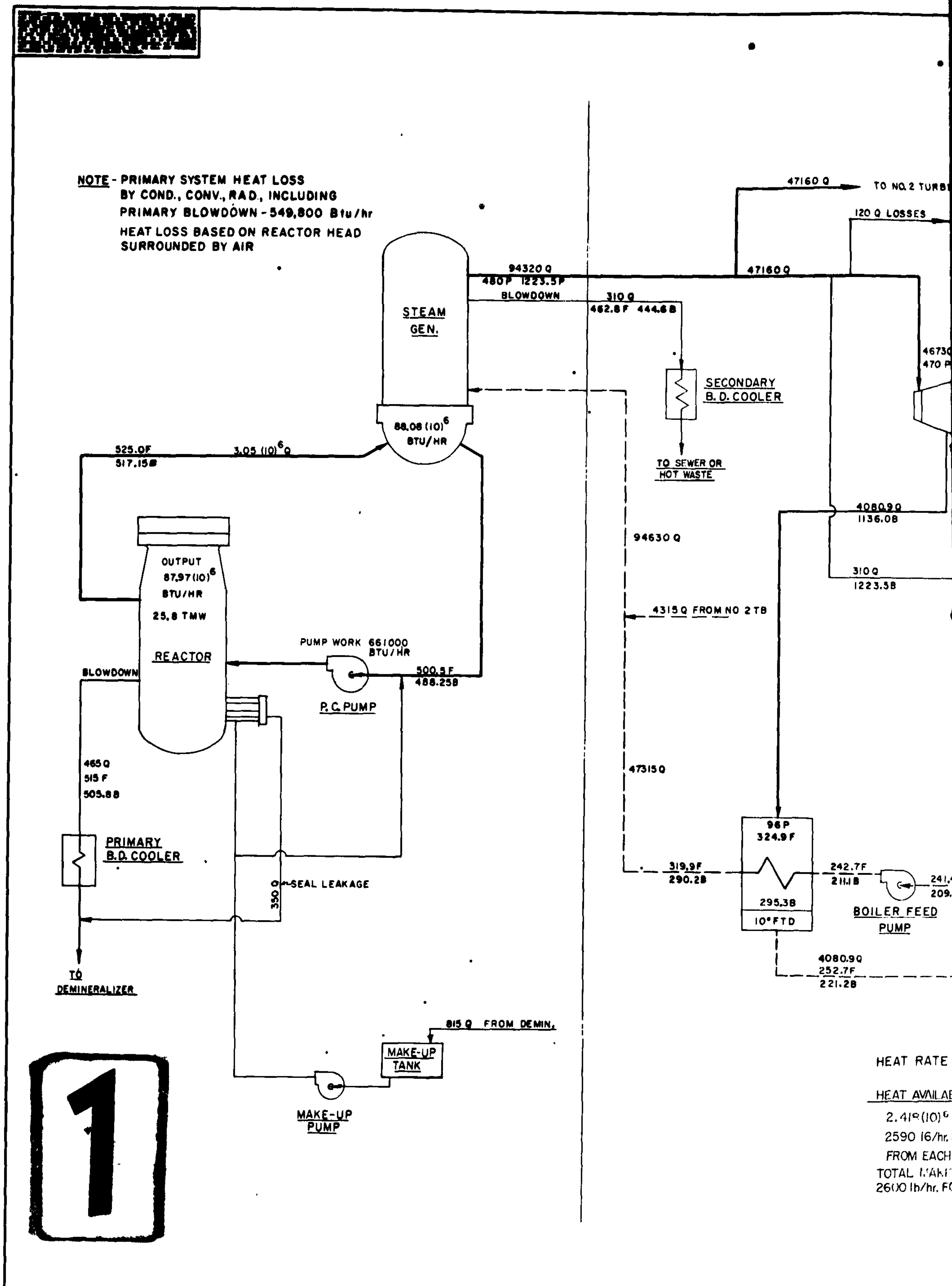
T. MEAD
D. M[illegible]
J. Angelo
[illegible]

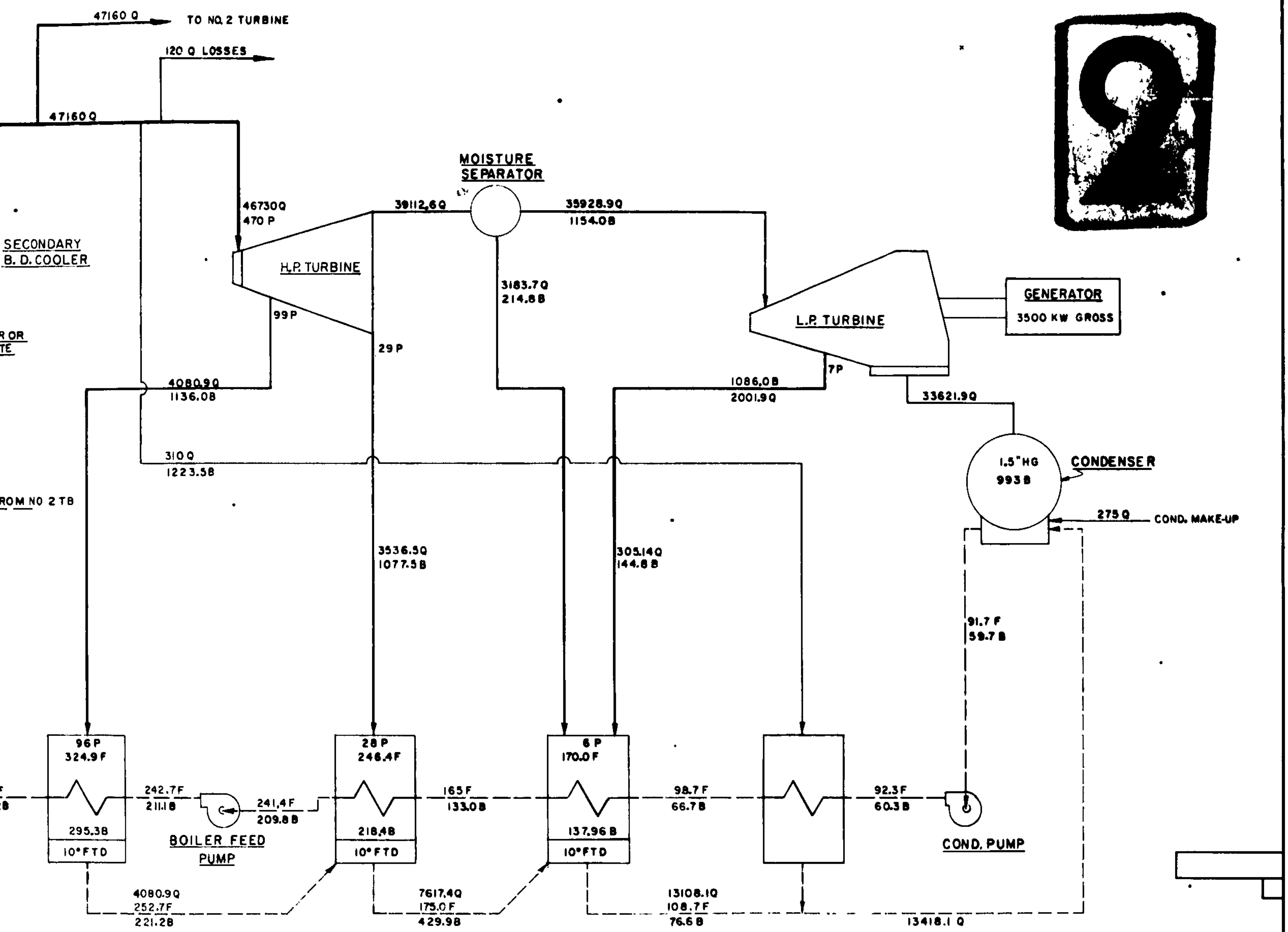
HEAT BALANCE
SM-2 6000eK.W
C3

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F M 11594-50

[redacted]


NOTE - PRIMARY SYSTEM HEAT LOSS
BY COND., CONV., RAD., INCLUDING
PRIMARY BLOWDOWN - 549,800 Btu/hr
HEAT LOSS BASED ON REACTOR HEAD
SURROUNDED BY AIR
STEAM GEN.
88.08 (10)^6 BTU/HR
94320 Q
480P 1223.5P
BLOWDOWN
310 Q
462.8F 444.6B
SECONDARY B.D. COOLER
TO SEWER OR HOT WASTE
47160 Q
TO NO. 2 TURB
120 Q LOSSES
47160 Q
525.0F
517.15B
3.05 (10)^6 Q
OUTPUT
87.97 (10)^6
BTU/HR
25.8 TMW
REACTOR
PUMP WORK 661000 BTU/HR
500.5F
488.25B
P.C. PUMP
BLOWDOWN
465 Q
515 F
505.88
PRIMARY B.D. COOLER
SEAL LEAKAGE
350 Q
TO DEMINERALIZER
815 Q FROM DEMIN.
MAKE-UP TANK
MAKE-UP PUMP
94630 Q
4315 Q FROM NO 2 TB
47315 Q
4080.9 Q
1136.0B
310 Q
1223.5B
96P
324.9F
319.9F
290.2B
295.3B
10°FTD
242.7F
211.1B
BOILER FEED PUMP
4080.9 Q
252.7F
221.2B
HEAT RATE
HEAT AVAILAB
2590 lb/hr.
FROM EACH
TOTAL MAKE
2600 lb/hr. FO
1



$$\text{HEAT RATE} = \frac{47040(1223.5) - 47315(290.2)}{3500} = 12550 \ \frac{\text{BTU}}{\text{KW-HR.}}$$

HEAT AVAILABLE FOR FLASH EVAPORATORS

$2.418(10)^6$ Btu/hr.

2590 lb/hr. STEAM FLOW

FROM EACH STEAM GENERATOR

TOTAL MAKE-UP CAPABILITY APPROX.

2600 lb/hr. FOR ENTIRE SYSTEM

LEGEND

Q = LB/HR

B = BTU/LB

P = PRESSURE

F = TEMP.

LIST OF MATERIAL

APPLICATION

SM-2 COMPLEX III

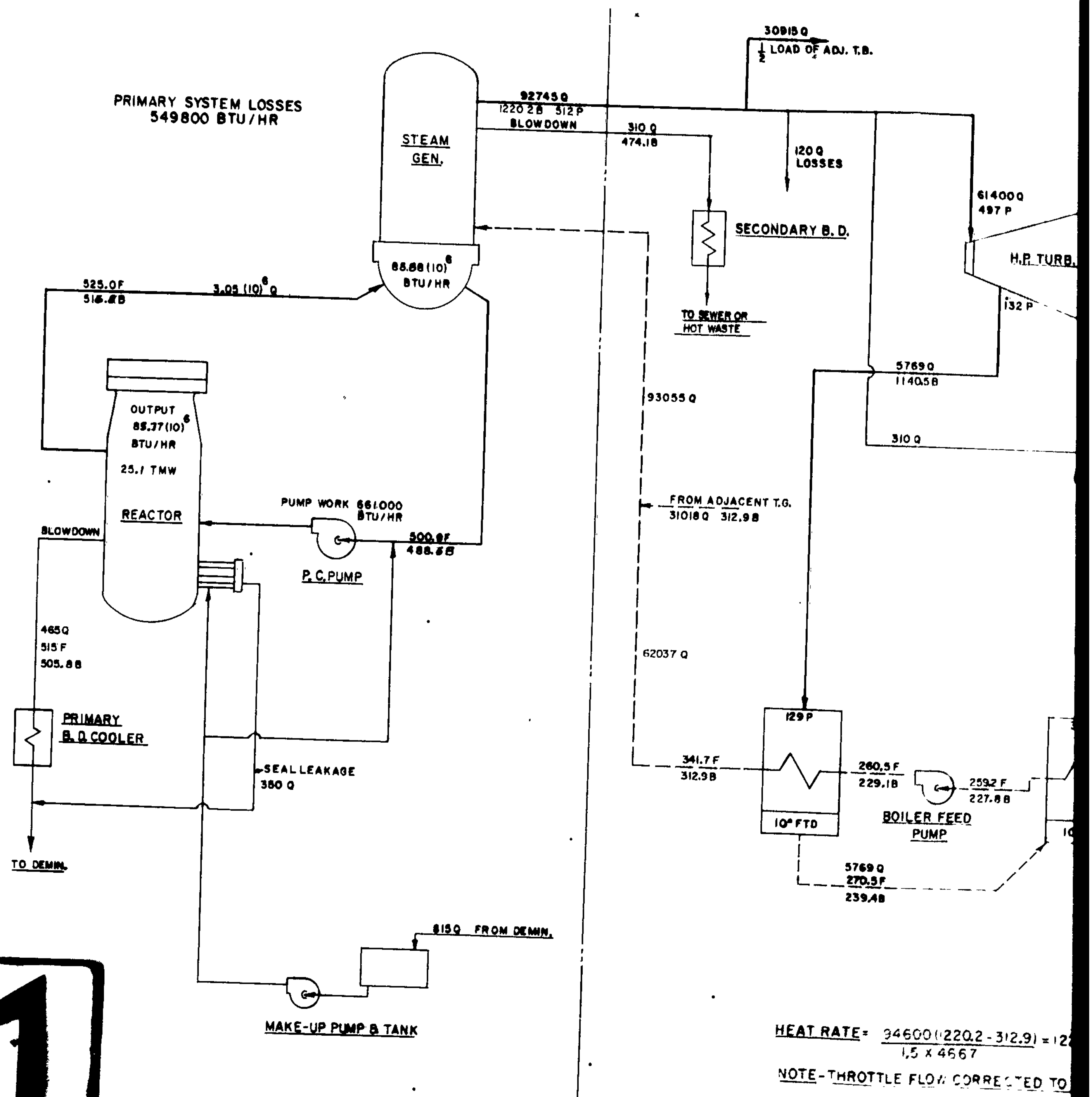
HEAT BALANCE

NORMAL LOAD COND.

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

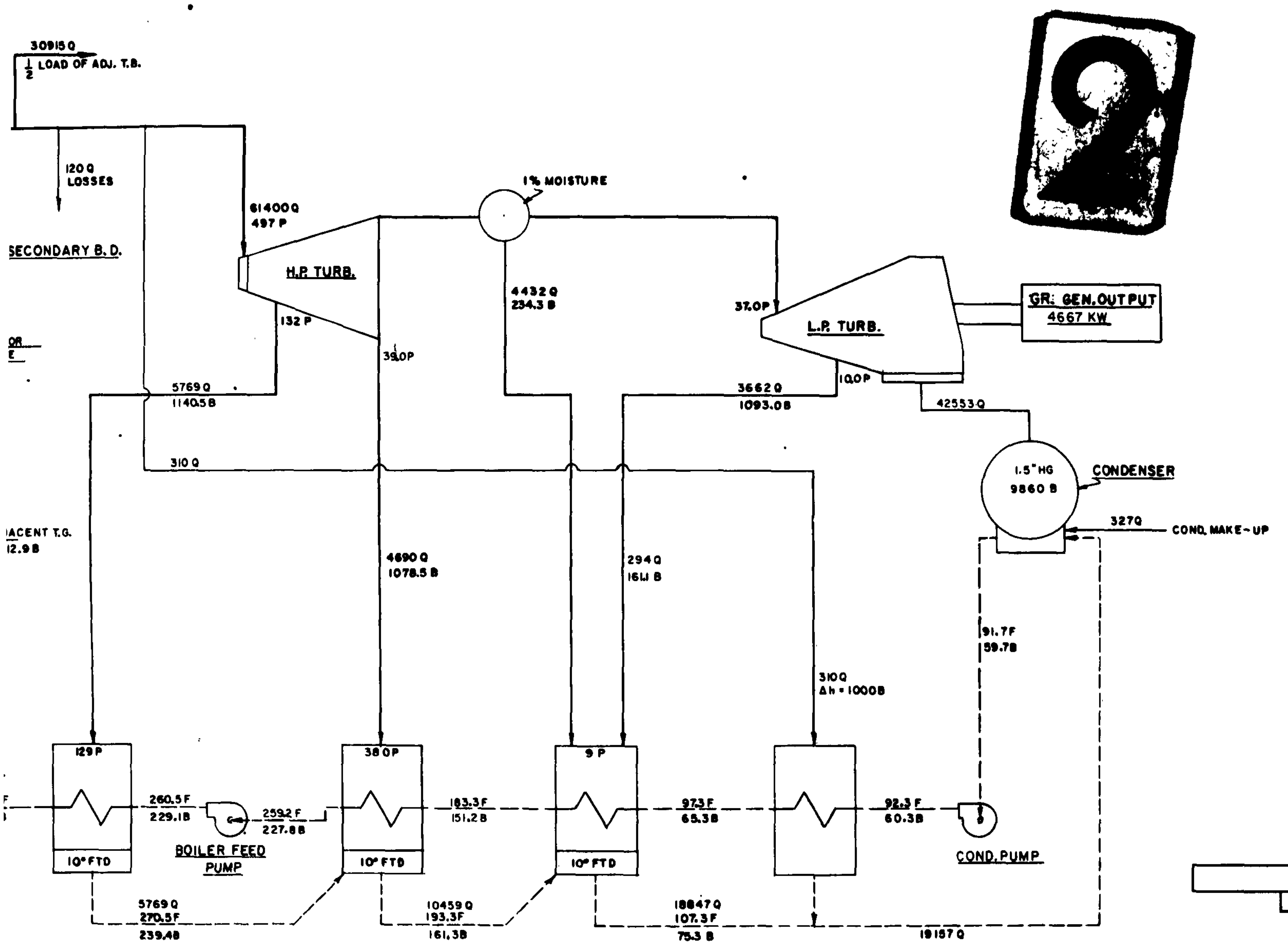
F M 11594-101

[redacted]



PRIMARY SYSTEM LOSSES
549800 BTU/HR
STEAM GEN.
85.88 (10)^6 BTU/HR
92745 Q
1220.2 B 512 P
BLOWDOWN
310 Q
474.1 B
30915 Q
1/2 LOAD OF ADJ. T.B.
120 Q
LOSSES
SECONDARY B. D.
TO SEWER OR HOT WASTE
61400 Q
497 P
H.P. TURB.
132 P
5769 Q
1140.5 B
310 Q
93055 Q
FROM ADJACENT T.G.
31018 Q 312.9 B
62037 Q
525.0 F
516.2 B
3.05 (10)^6 Q
OUTPUT
85.37 (10)^6
BTU/HR
25.1 TMW
REACTOR
PUMP WORK 661,000 BTU/HR
P. C. PUMP
500.9 F
488.5 B
BLOWDOWN
465 Q
515 F
505.8 B
PRIMARY B. D. COOLER
SEAL LEAKAGE
380 Q
TO DEMIN.
129 P
341.7 F
312.9 B
10° FTD
260.5 F
229.1 B
259.2 F
227.8 B
BOILER FEED PUMP
5769 Q
270.5 F
239.4 B
815 Q FROM DEMIN.
MAKE-UP PUMP & TANK
HEAT RATE = 94600 (1220.2 - 312.9) = 12
1.5 x 4667
NOTE - THROTTLE FLOW CORRECTED TO

1



LEGEND
Q = LB/HR
B = BTU/LB
P = PRESSURE
F = TEMP

HEAT RATE = $\frac{94600\,(1220.2 - 312.9)}{1.5 \times 4667} = 12250$ BTU/KW-HR

NOTE - THROTTLE FLOW CORRECTED TO 94600 Q FOR 4% MECH. & ELEC. LOSSES

LIST OF MATERIAL

APPLICATION

SM-2 COMPLEX III
HEAT BALANCE
3 TURBINES AT FULL LOAD

M 11594-102

4.0 PRIMARY LOOP FLOW

4.1 DESCRIPTION

Drawing M 11594-49 is a schematic representation of the piping in the reactor and primary auxiliary sections of the SM-2 plant. The primary coolant circulating loop is described here; the auxiliary systems are covered in the sections dealing with auxiliary components.

The function of the primary coolant loop is to transfer heat generated in the reactor core to the steam generator. This is accomplished by heat transfer to and from the light water coolant which is circulated in the primary coolant loop.

The primary coolant loop consists of the reactor vessel, steam generator, pressurizer, primary circulating pump, and interconnecting piping. Primary coolant is discharged at a rate of 7800 gpm from the primary circulating pump through 14 in. diam stainless steel piping to the reactor vessel. The coolant enters the reactor vessel at a temperature of 500.5°F, and makes two passes through the reactor core, where its temperature is raised to 525.5°F. It is then discharged through 14 in. diam piping to the tube side of the vertical steam generator. In the steam generator, the coolant passes through a U-type tube bundle where it gives up heat at the full load rate of 90.5×10^6 Btu/hr to generate superheated steam. The coolant then leaves the steam generator and returns through a 16 in. stainless steel elbow to the primary circulating pump, thus completing the cycle.

A pressurizer maintains a pressure of 2000 psia on the system during operation. It is connected to the 14 in. piping in the section between the steam generator and the reactor vessel inlet. The connecting line is a 3 in. pipe which is essentially a dead leg. The only flow through this pipe is that which is induced by load fluctuations.

A 2 in. line which connects the pressurizer's vapor chamber with the 14 in. piping downstream of the primary circulating pump is provided to facilitate accelerated cooldown of the pressurizer during a plant shutdown. A manually operated globe valve is provided in this line and is normally closed. The valve is opened only when system cooldown is desired.

The primary coolant loop is protected by two 1-1/2 by 2-1/2 in. relief valves mounted on the pressurizer. As the system design pressure is 2200 psia, the relief valves are set to open at this pressure.

Provision for constant makeup and blowdown is made to maintain purity of the circulating fluid. The system is blown down from the steam generator channel and the reactor vessel. Makeup water enters the system at a penetration in the piping downstream of the primary circulating pump.

A fill line and vent connection are provided for the motor cavity of the primary circulating pump to insure that proper motor bearing lubrication is provided prior to operation.

The primary piping loop is all-welded construction, and will be shop fabricated. The only field connections will be those for the pressurizer.

Two valves, capped with blind flanges, are provided for connection of temporary piping used during decontamination of the system.

4.2 PIPE SIZES AND MATERIALS

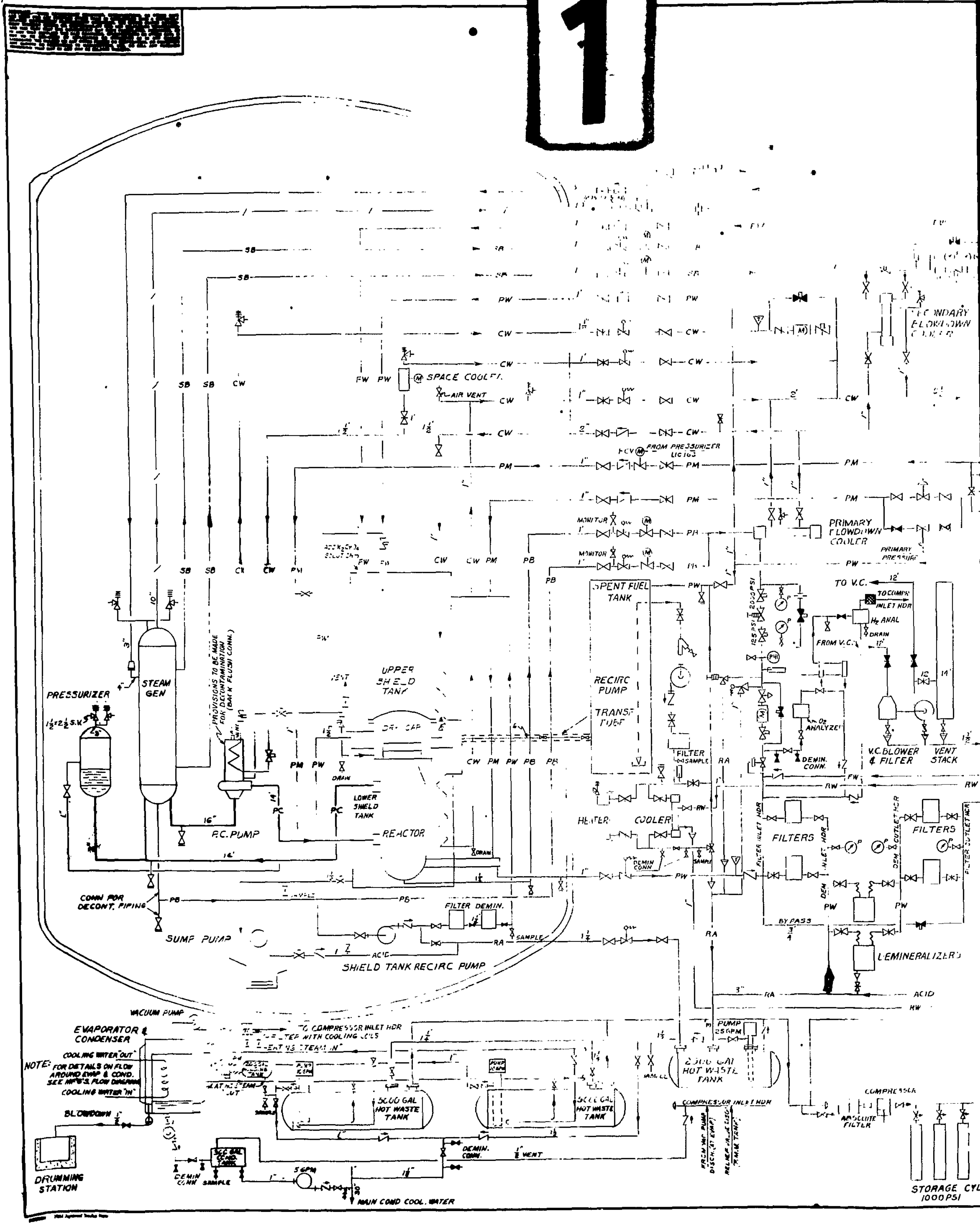
Line Description	Size In.	Thk. Sch.	Spec. ASTM	Flow Medium	Line Flow From	Line Flow To	Oper. @ Full Load Press. Psia	Oper. @ Full Load Temp. ^oF	Insulation	Remarks
Primary Coolant	14	140	A376 T304	Water	Reactor	Steam Gen.	2000	528	yes	Insulation for Protection
Primary Coolant	16	140	A376 T304	Water	Steam Gen.	Prim. Pump	2000	503	yes	
Primary Coolant	14	140	A376	Water	Prim. Pump	Reactor	2000	514	yes	
Pressurizer - Equal.	2	160	A376 T304	Water	Line to Steam Gen.	Pressurizer	2000	636	yes	Insulation for Protection
Pressurizer - Cooldown	2	160	A376 T304	Water	Line to Steam Gen.	Pressurizer	2000	512	yes	
Primary Makeup	1-1/2	80	A376 T304	Water	PMU Tank	Pumps	30	100	no	
Primary Makeup	1	80	A376 T304	Water	Pumps	Prim. Circuit	2100	100	no	
Primary Makeup	1	80	A376 T304	Water	Header	Control Rod Seals	2175	100	no	
Main Steam	10	80	A106 Gr. B	Steam	Steam Gen	Turbine	480	486^oF TT	yes	Insulation for Protection
Feedwater	3	80	A106 Gr. B	Water	Boiler Feed Pumps	Steam Gen.	560	340	yes	
Condensate	1	80	A106 Gr. B	Water	Main Steam	Cond. Well	480	486	yes	

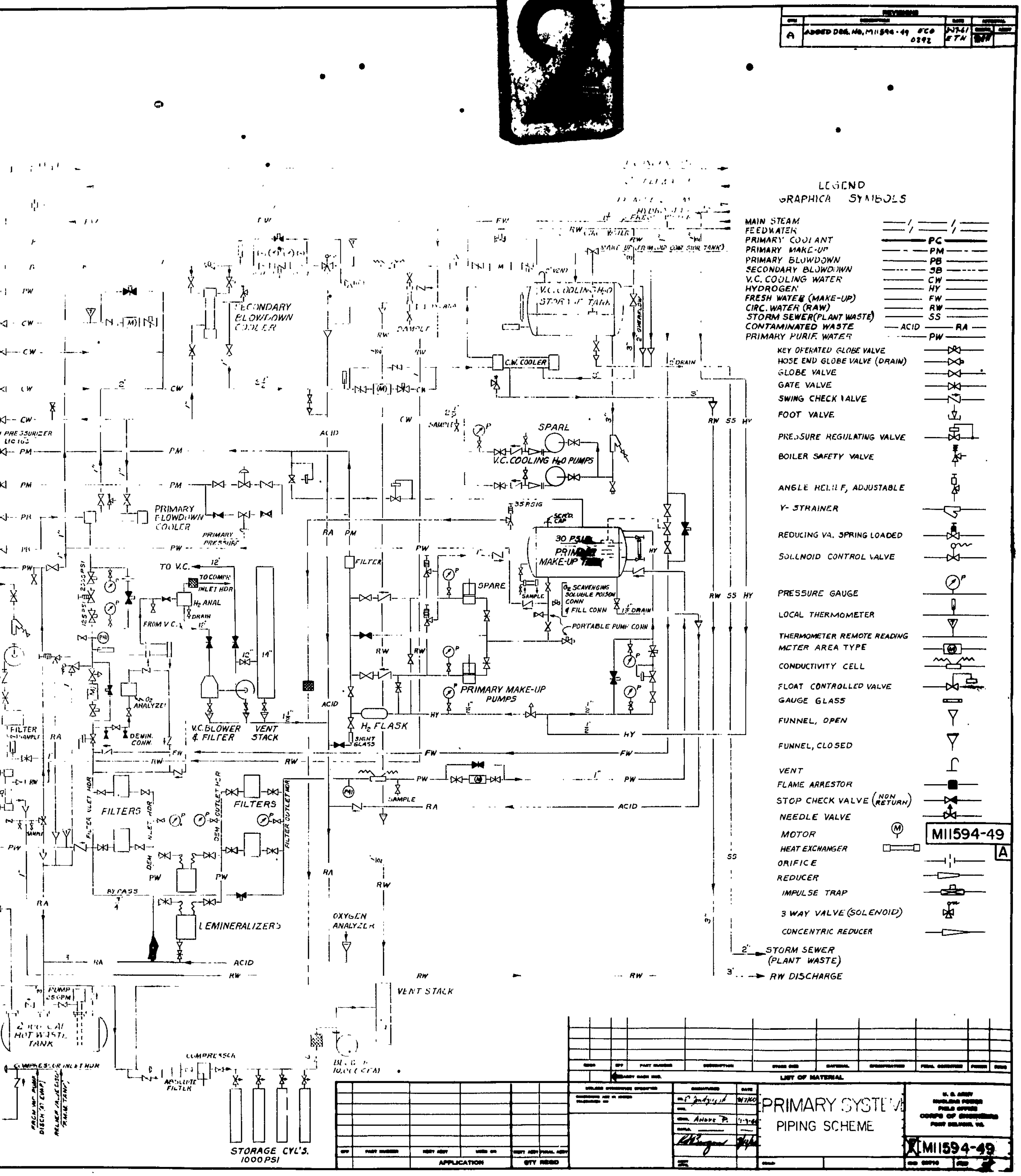
Line Description	Size In.	Thk. Sch.	Spec. ASTM	Flow Medium	Line Flow From	Line Flow To	Oper. @ Full Load Press. Psia	Oper. @ Full Load Temp. ^{o}F	Insulation	Remarks
Primary Blowdown	1"	160	A376 T304	Water	Reactor	PBD Cooler	2000	518	yes	Insulation for Protection
	1	160	A376 T304	Water	Cooler	Press. Valve	2000	110	no	
	1"	160	A376 T304	Water	Stm. Gen.	PBD Cooler	2000	518	yes	Insulation for Protection
	1	80	A376 T304	Water	Press Valve	Demin. & Storage	125	110	no	
Secondary Blowdown	1	80	A106 Gr. B	Water	Stm. Gen.	SBD Cooler	480	463	yes	Insulation for Protection
	1	80	A106 Gr. B	Water	SBD Cooler	Press Valve	480	110	no	
	1	80	A106 Gr. B	Water	Press Valve	Drain	110	110	no	
V. C. Cooling	3"	40	A106 Gr. B	Water	Tank	Pumps	30	100	no	
	2-1/2"	40	A106 Gr. B	Water	Pumps	Header (Skid)	55	100	no	
	1"	80	A106 Gr. B	Water	Main Header	PBD Cooler	55	100	no	
	1"	80	A106 Gr. B	Water	Main Header	SBD Cooler	55	100	no	
	2"	40	A106 Gr. B	Water	Main Header	Vapor Cont.	55	100	no	
	1-1/2"	80	A106 Gr. B	Water	Main Header	P. C. Pump	55	100	no	
	1	80	A106 Gr. B	Water	Main Header	Shield Tank Coil	55	100	no	
	1	80	A106 Gr. B	Water	Main Header	Space Cooler	55	100	no	
	2	40	A106 Gr. B	Water	Equip. in V. C.	Collect Head	40	130	no	

Line Description	Size In.	Thk. Sch.	Spec. ASTM	Flow Medium	Line Flow From	Line Flow To	Oper. @ Full Load Press. Psia	Oper. @ Full Load Temp. °F	Insul-ation	Remarks
	2-1/2"	40	A106 Gr. B	Water	Collect Head	Skid Head	40	130	no	
	2-1/2"	40	A106 Gr. B	Water	Skid Head	C. W. Cooler	40	130	no	
	2-1/2"	40	A106 Gr. B	Water	C. W. Cooler	Tank	35	130	no	
Service (raw) Water	3"	40	A106 Gr. B	Water	Header	Cooler	18	70	no	
	3"	40	A106 Gr. B	Water	Cooler	Discharge	18	100	no	
	1/2"	40	A106 Gr. B	Water	Header	Ox. Anal.	18	70	no	
	1/2"	40	A106 Gr. B	Water	Ox. Anal.	Discharge	18	70	no	
	1"	40	A106 Gr. B	Water	Header	S. F. T. Cooler	18	70	no	
	1"	40	A106 Gr. B	Water	S. F. T. Cooler	Discharge	18	100	no	
Purification - Shield Tank	1-1/4	80	A376 T304	Water	Recirc. Pump	Shield Tank	27	100	no	
	1-1/2	80	A376 T304	Water	Shield Tank	Pump	ATM.	100	no	
	1"	80	A376 T304	Water	Header	Shield Tank	30	100	no	
Spent Fuel Tank	1"	80	A376 T304	Water	Header	S. F. Tank	30	100	no	
	1-1/4"	80	A376 T304	Water	Pump	Filter & Cooler	43	100	no	
	1-1/4"	80	A376 T304	Water	Filter-Cooler	S. F. Tank	35	100	no	
	1-1/2"	80	A376 T304	Water	S. F. Tank	Pump	ATM.	100	no	

Line Description	Size In.	Thk. Sch.	Spec. ASTM	Flow Medium	Line Flow From	Line Flow To	Oper. @ Full Load Press. Psia	Oper. @ Full Load Temp. °F	Insulation	Remarks
Fresh Water	1-1/2	40	A106 Gr. B	Water	Header	V. C.	120	100	no	
	1	40	A106 Gr. B	Water	V. C.	Shield Tank	120	100	no	
	1	40	A106 Gr. B	Water	Header	Filter Inlet	120	100	no	
Contaminated Waste	1"	80	A376 T304	Water	PBD Return	Hot Waste Tank	ATH	110°F	no	
	1"	80	A376 T304	Water	SBD Return	Hot Waste Tank	ATH	110°F	no	
	1-1/2"	80	A376 T304	Water	PMU Tank	Hot Waste Tank	ATH	110°F	no	
	1-1/4"	80	A376 T304	Water	Shield Tank Recirc.	Hot Waste Tank	ATH	110°F	no	
Storm Sewer	2"	40	A106 Gr. B	Water	Tank	Plant Waste	ATH	70	no	
Hydrogen	1/2	80	A376 T304	H_2	Header	PMU Tank	2250	70	no	
	1/2	80	A376 T304		Header	PMU Pump Discharge	2250	70	no	
Air Ventilation	12	10	A106 Gr. B	Air	Fan	12" Valve	B H_2O	100	no	
	12	40	A106 Gr. B	Air	12" Valve	V. C.	B H_2O	100	no	
	12	40	A106 Gr. B	Air	V. C.	12" Valve	B H_2O	100	no	
	12	10	P106 Gr. B	Air	12" Valve	Filter	B H_2O	100	no	

1


PRESSURIZER
STEAM GEN
PROVISIONS TO BE MADE FOR DECONTAMINATION (BACK FLUSH CONN.)
R.C. PUMP
UPPER SHIELD TANK
LOWER SHIELD TANK
REACTOR
SPACE COOLER
AIR VENT
FROM PRESSURIZER LIC 163
MONITOR
SPENT FUEL TANK
RECIRC PUMP
FILTER
SAMPLE
HEATER
COOLER
DEMIN CONN
PRIMARY BLOWDOWN COOLER
SECONDARY BLOWDOWN COOLER
TO V.C.
TO COMPR INLET HDR
H_2 ANAL
DRAIN
FROM V.C.
O_2 ANALYZER
DEMIN. CONN.
V.C. BLOWER & FILTER
VENT STACK
FILTERS
FILTER INLET HDR
FILTER OUTLET HDR
DEM INLET HDR
DEM OUTLET HDR
BY PASS
DEMINERALIZERS
ACID
CONN FOR DECONT. PIPING
SUMP PUMP
FILTER DEMIN.
SHIELD TANK RECIRC PUMP
VACUUM PUMP
EVAPORATOR & CONDENSER
NOTE: FOR DETAILS ON FLOW AROUND EVAP & COND. SEE MFG'S. FLOW DIAGRAM
COOLING WATER "OUT"
COOLING WATER "IN"
BLOWDOWN
DRUMMING STATION
5000 GAL HOT WASTE TANK
500 GAL COND. TANK
5 GPM
DEMIN CONN
SAMPLE
MAIN COND COOL. WATER
DEMIN. CONN.
VENT
PUMP 25 GPM
2500 GAL HOT WASTE TANK
COMPRESSOR INLET HDR
COMPRESSOR
ABSOLUTE FILTER
STORAGE CYL'S 1000 PSI
CW
PW
PM
PB
PC
SB
RA
RW
FW



A
ADDED DWG. No. M11594-49 FCO 0292
LEGEND
GRAPHICAL SYMBOLS
MAIN STEAM
FEEDWATER
PRIMARY COOLANT — PC
PRIMARY MAKE-UP — PM
PRIMARY BLOWDOWN — PB
SECONDARY BLOWDOWN — SB
V.C. COOLING WATER — CW
HYDROGEN — HY
FRESH WATER (MAKE-UP) — FW
CIRC. WATER (RAW) — RW
STORM SEWER (PLANT WASTE) — SS
CONTAMINATED WASTE — ACID — RA
PRIMARY PURIF. WATER — PW
KEY OPERATED GLOBE VALVE
HOSE END GLOBE VALVE (DRAIN)
GLOBE VALVE
GATE VALVE
SWING CHECK VALVE
FOOT VALVE
PRESSURE REGULATING VALVE
BOILER SAFETY VALVE
ANGLE RELIEF, ADJUSTABLE
Y- STRAINER
REDUCING VA. SPRING LOADED
SOLENOID CONTROL VALVE
PRESSURE GAUGE
LOCAL THERMOMETER
THERMOMETER REMOTE READING
METER AREA TYPE
CONDUCTIVITY CELL
FLOAT CONTROLLED VALVE
GAUGE GLASS
FUNNEL, OPEN
FUNNEL, CLOSED
VENT
FLAME ARRESTOR
STOP CHECK VALVE (NON RETURN)
NEEDLE VALVE
MOTOR
HEAT EXCHANGER
ORIFICE
REDUCER
IMPULSE TRAP
3 WAY VALVE (SOLENOID)
CONCENTRIC REDUCER
M11594-49
A
SECONDARY BLOWDOWN COOLER
V.C. COOLING H2O STORAGE TANK
C.W. COOLER
SPARE
V.C. COOLING H2O PUMPS
PRIMARY BLOWDOWN COOLER
PRIMARY PRESSURE
TO V.C.
TO COMPR. INLET HDR
H2 ANAL
FROM V.C.
O2 ANALYZER
DEMIN. CONN.
V.C. BLOWER & FILTER
VENT STACK
30 PSIG PRIMARY MAKE-UP TANK
35 PSIG
O2 SCAVENGING SOLUBLE POISON CONN
FILL CONN
PORTABLE PUMP CONN
SAMPLE
DRAIN
FILTER
PRIMARY MAKE-UP PUMPS
H2 FLASK
SIGHT GLASS
FILTERS
FILTER INLET HDR
FILTER OUTLET HDR
DEMINERALIZERS
BY PASS
OXYGEN ANALYZER
VENT STACK
STORM SEWER (PLANT WASTE)
RW DISCHARGE
HOT WASTE TANK
COMPRESSOR
ABSOLUTE FILTER
STORAGE CYL'S. 1000 PSI
LIST OF MATERIAL
APPLICATION
QTY REQD
PRIMARY SYSTEM
PIPING SCHEME
U. S. ARMY
CORPS OF ENGINEERS
M11594-49

5.0 PLANT THERMAL AND HYDRAULIC ANALYSIS

5.1 CONTRACT REQUIREMENTS

Contract No. DA-44-009-ENG-3506, Exhibit B, required that a system development analysis be conducted on the pressurizer and controls. This work was to be done in two parts:

1. A literature search and analysis of the rate of condensation on metal walls and liquid interface, the added surface and thermal mass to back up the surface; and pressure and liquid level programming.

2. Incorporation of the above in an analog model for the selection of the size of the SM-2 pressurizer.

This development analysis is completed and has been reported in APAE Memo No. 266. (1).

Under Exhibit C of Contract No. DA-44-192-ENG-7, this section of the report covers the following items in the primary system:

1. Primary loop pressure drop

2. Pressurizer

3. Transient design analysis

5.2 DESIGN INFORMATION, PLANT CONSTANTS

Plant Performance

Thermal power developed in reactor	MW	28
Electric power generated	KW	7000

Reactor Dimensions

Height of active core	In.	22.0
Dimensions of fuel element coolant passages (inches)		
	Fixed	Control Rod
Length	23.5	23.0
Thickness	0.123	0.123
Distance between side plates	2.789	2.544

Thermal Data of Reactor

Coolant inlet temperature at reactor	^oF	500
Coolant outlet temperature at reactor	^oF	525
Properties of coolant @ 500^oF and 2000 psia		
Density	lb/ft^3	49.5
Viscosity	lb/hr ft	0.260
Thermal conductivity	Btu/hr ft^oF	0.3562
Specific heat	Btu/hr ^oF	1.154
Properties of coolant @ 80^oF and 2000 psia		
Density	lbs/ft^3	62.55
Viscosity	lbs/hr ft	2.083

Hydraulic Data of Reactor at Full Power

Operating pressure in reactor	psia	2000
Primary loop pressure drop	ft H_2O	113.9
Number of reactor coolant passes		2
Coolant flow in core passages/element	gpm	
1st pass -stationary elements		218
2nd pass -stationary elements		316
Control rod		331
Coolant requirements	ft^3/sec	17.38
	gpm	7800

Piping

Type, 304 ss

Nominal pipe sizes, schedule 140

Between pump and reactor, reactor and steam generator	in.	14
Between steam generator and pump	in.	16
Inside diameters	ft (14")	0.9583
	ft (16")	1.0937
Cross-sectional areas	ft^2 (14")	0.7213
	ft^2 (16")	0.9394

Reference Drawings

Primary loop	M 11594-57
Pressurizer	M 11594-40

5.3 PRIMARY LOOP PUMPING REQUIREMENTS

5.3.1 Nomenclature

A	Pipe cross-sectional area, ft^2
C	bend loss coefficient, dimensionless
d, D	pipe diameter, ft
f	friction factor, dimensionless
g	acceleration due to gravity = 32.2 ft/sec^2
h	head loss, ft of N_2O
K	loss coefficient due to curvature of pipe bend
K'	overall loss coefficient
l	equivalent length of pipe, ft
L. R.	long radius (referring to pipe bend), ft
P	pressure drop, ft of H_2O
Q	fluid flow, gpm
r	radius of pipe bend, ft
N_{RE}	Reynolds' Number = $\frac{\rho VD}{\mu}$ dimensionless
S. R.	short radius (referring to pipe bend), ft
V	fluid velocity, ft/sec
ρ	density, lbs/ft^2
μ	viscosity lbs/hr ft
x	as subscript; unknown quantity
1	as subscript; know quantity

5.4 PRESSURE DROP CALCULATIONS

5.4.1 Calculation of Primary Loop Piping Losses

5.4.1.1 Reactor to Steam Generator Section

A. Frictional Loss

Bulk fluid temperature assumed to be at $525^{o}F$.

Total equivalent length of piping = 127.43 in. (14" sch. 140)

$$V_{(14'' \text{ Sch } 140)} = \frac{Q}{A} = \frac{7800 \times .002228}{0.7213} = 24.09 \text{ ft/sec}$$

$$N_{Re\ (525^{o}-14'')} = \frac{\rho VD}{\mu} = \frac{(48.0)\ (24.09)\ (3600)\ (.9583)}{(.2465)} = 16.18 \times 10^{6}$$

f = 0.008 (AEC Handbook[1])

$$h = \frac{fl}{d}\ \frac{V^2}{2g} = \left[\frac{(.008\ (\frac{127.32}{12})}{.9583}\right]\ \left[\frac{(24.09)^2}{64.4}\right] = 0.80 \text{ ft } H_2O$$

B. Kinetic Losses

The section under consideration includes 1-28-1/2^{o} S.R. elbow, 1-45^{o} L.R. elbow, 1-90^{o} S.R. elbow and 1-90^{o} L.R. elbow.

$$\frac{r}{d} \text{ ratio for S.R.} = \frac{14}{11.5} = 1.22$$

$$\frac{r}{d} \text{ ratio for L.R.} = \frac{21}{11.5} = 1.83$$

The following coefficients were taken from reference 2.

K (ratio = 1.22) = 0.29

K (ratio = 1.83) = 0.21

for 28-1/2^{o}, bend coefficient C = 0.41

for 45^{o}, bend coefficient C = 0.62.

for 90^{o}, bend coefficient C = 1.00

Total loss coefficient, K', is defined as equal to (K) (C).

K' (28-1/2°, S.R.) = (.29) (.41) = 0.12

K' (45°, L.R.) = (.21) (.62) = 0.13

K' (90°, S.R.) = (.29) (1.00) = 0.29

K' (90°, L.R.) = (.21) (1.00) = $\frac{0.21}{0.75}$

$$\frac{V^2}{2g} = \frac{(24.09)^2}{64.4} = 9.01 \text{ ft}$$

Total kinetic loss = 0.75 (9.01) = 6.76 ft H_2O

5.4.1.2 Steam Generator to Pump Loss

A. Frictional Loss

Bulk fluid temperature assumed to be at 500°F.

Total equivalent length of piping = 73.55 in (16" sch 140)

$$V_{(16'' \text{ sch } 140)} = \frac{Q}{A} = \frac{7800 \times 0.002228}{135.32/144} = 18.49 \text{ ft/sec}$$

$$N_{RE\ (500°F\ -\ 16'')} = \frac{(49.5)(18.49)(3600)(1.0937)}{0.260} = 13.86 \times 10^6$$

f = .008

$$h = \left[\frac{(.008)\left(\frac{73.55}{12}\right)}{1.0937}\right]\left[\frac{(18.49)^2}{64.4}\right] = 0.24 \text{ ft } H_2O$$

B. Kinetic Losses

The section under consideration includes 1-45° L.R. elbow and 1-90° L.R. elbow.

Total loss coefficient

K' (45° L.R.) = (.21) (.62) = 0.13

K' (90° L.R.) = (.21) (1.00) = $\frac{0.21}{0.34}$

$\frac{r}{d}$ ratio for L.R. = $\frac{24}{13.12}$ = 1.83 K = 0.21

$$\frac{V^2}{2g} = \frac{(18.49)^2}{64.4} = 5.31 \text{ ft.}$$

Total kinetic loss = 0.34 (5.31) = 1.81 ft.

5.4.1.3 Pump to Reactor Loss

A. Frictional Loss

Bulk fluid temperature assumed to be at $500^{\circ}F$

Total equivalent length of piping = 87.87 in (14" sch 140)

$V_{(14'' \text{ sch } 140)} = 24.09$ ft/sec

$$N_{RE\ (500^{\circ}-14)} = \frac{(49.5)(24.09)(3600)(.9583)}{0.260} = 15.82 \times 10^{6}$$

$$f = 0.008 \text{ (Ref. 1)}$$

$$h = \left[\frac{(.008)\left(\frac{87.87}{12}\right)}{0.9583}\right]\left[\frac{(24.09)^2}{64.4}\right] = 0.55 \text{ ft } H_2O$$

B. Kinetic Losses

The section under consideration includes 1-20° S.R. elbow and 2-90° S.R. elbows.

Total loss coefficient

$K' (20^{\circ}, \text{S.R.}) = (.29)(.30) = 0.09$

$K' (90^{\circ}, \text{S.R.}) = (2)(.29)(1.00) = \frac{0.58}{0.67}$

$$\frac{V^2}{2g} = 9.01 \text{ ft}$$

Total kinetic loss = 0.67 (9.01) = 6.04 ft H_2O

5.4.1.4 Summary, Primary Piping Losses, (7800 gpm)

Loop Section	Description of Piping [Nominal Pipe Diam. in Parenthesis]	Equivalent Piping Lgt. In.	Pressure Drop, Ft of H_2O Hot Conditions Frictional Loss	Hot Conditions Kinetic Loss	Cold Conditions Frictional Loss	Cold Conditions Kinetic Loss
Reactor to Steam Generator (525^oF)	22-1/2" + 26-1/2"(14")	49.00				
	28-1/2^o S.R. ell (14")	6.96		1.08		1.08
	45^o L.R. ell (14")	16.49		2.177		1.17
	90^o S.R. ell (14")	21.99		2.62		2.62
	90^o L.R. ell (14")	32.99		1.89		1.89
Totals		127.43	0.80	6.76	1.23	6.76
Steam Generator to Pump (500^oF)	17" (16")	17.00		----		----
	45^o L.R. ell (16")	37.70		0.69		0.69
	90^o L.R. ell (16")	18.85		1.12		1.12
Totals		73.55	0.24	1.81	0.57	1.81
Pump to Reactor (500^oF	18" + 21" (14")	39.00		----		----
	20^o S.R. ell (14")	4.89		0.80		0.80
	90^o S.R. ell (14")	21.99		2.62		2.62
	90^o S.R. ell (14")	21.99		2.62		2.62
Totals		87.87	0.55	6.04	0.92	6.04
Total Piping Losses			1.59	14.61	2.72	14.61

5.4.2 Components Analyzed Elsewhere

5.4.2.1 Reactor Losses

The following values have been calculated for the reactor core and vessel pressure drops at 7800 gpm, hot condition.[2]

A. 1st Pass, Shield and Vessel

1.	Frictional loss (fuel section, shields)	= 9.2 ft H_2O
2.	Kinetic loss (nozzles, bottom and top plate)	= 15.8 ft H_2O
	Total loss	= 25.0 ft H_2O

B. 2nd Pass

1.	Frictional loss (plates and absorber)	= 10.5 ft H_2O
2.	Kinetic loss (inlet and outlet losses)	= 15.2 ft H_2O
	Total loss	= 25.7 ft H_2O

5.4.2.2 Steam Generator Loss

The overall nozzle to nozzle steam generator primary side pressure drop is 12.0 psi at 7800 gpm and under hot flow conditions at $512^{\circ}F$ the average primary temperature, ρ, is 49.5 lbs/ ft^3.

$$\text{Head in feet of water} = \frac{\Delta P \text{ in p. s. f.}}{\rho \text{ in lb/ft}^3}$$

$$\Delta P = 12.0 \times \frac{144}{49.5} = 34.9 \text{ ft } H_2O$$

This loss of 12.0 psi was obtained emprically from curves. No breakdown of this loss into frictional and kinetic losses is thus available. However, a previous analysis of a similar SM-2 steam generator[3] indicates for a total ΔP of 55.2 ft H_2O, 43.7 ft is frictional, 11.5 ft is kinetic loss, or 79 percent frictional, 21 percent kinetic. This proportion was assumed in the present analysis.

Frictional loss = 34.9 x 0.79 = 27.6 ft

Kinetic loss = 34.9 x 0.21 = 7.3 ft

5.4.3 Conclusions

5.4.3.1 Results at Operating Temperature

The calculated pressure drop around the complete SM-2 primary loop at the required flow rate of 7800 gpm is 103.5 ft H_2O at reactor operating temperature. The recommended specification head requirement for the primary loop pump is 113.9 ft H_2O at 500^{o}F which provides a 10 percent margin over the directly calculated value.

The following tabulation shows the magnitude of the pressure drops of each component (including piping) in the primary loop at arbitrarily selected flow rates of 6500, 7000, 7500 and 8000 gpm. Figure 5-1 gives a graphic representation.

5.4.3.2 Results at Startup Temperature

The anticipated startup pressure drop is based on an increased friction factor at lower water temperatures due to a decreased Reynold's number.

The calculated pressure drop at the required flow rate at 7800 gpm is 123.0 ft H_2O at a normal startup temperature of 80^{o}F.

ANALYSIS OF PRIMARY LOOP PRESSURE DROPS AT VARYING CONDITIONS OF TEMPERATURE AND FLOW

(Pressure Drops in Ft of H_2O)

	6500 GPM						7000 GPM						7500 GPM						8000 GPM					
	Cold			Hot			Cold			Hot			Cold			Hot			Cold			Hot		
	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.	ΔP_{FR}	ΔP_K	Tot.
1st Pass Shield, Vessel	9.1	11.0	20.1	6.6	11.0	17.6	10.4	12.7	23.1	7.6	12.7	20.3	11.8	14.6	26.4	8.6	14.6	23.2	13.3	16.6	29.9	9.6	16.6	26.2
2nd Pass (Control Rods)	11.7	11.4	23.1	8.4	11.4	19.8	13.2	13.4	26.6	9.6	13.4	22.9	14.9	14.8	29.7	10.8	14.8	25.6	17.0	17.2	31.2	12.3	17.3	29.6
Piping	1.5	10.1	11.7	1.2	10.1	11.3	1.7	11.8	13.5	1.4	11.8	13.2	2.0	13.5	15.5	1.5	13.5	15.0	2.2	15.4	17.6	1.7	15.4	17.1
Steam Generator	27.4	5.1	32.5	19.8	5.1	24.9	31.3	5.9	32.7	22.7	5.9	28.6	35.5	6.8	42.3	25.7	6.8	32.5	39.9	7.7	47.6	28.9	7.7	36.6
Total			87.4			73.6			100.4			85.0			113.9			96.2			129.3			109.5

NOTES: ΔP_{FR}: That portion of pressure drop due to frictional losses (length of pipe)

ΔP_K: That portion of pressure drop due to kinetic losses (pipe bends)

5.5 FLOW COASTDOWN BEHAVIOR

5.5.1 Description

Upon failure of the primary coolant pump, the circulation of the coolant drops off rapidly as frictional resistance to flow through the loop dissipates the available kinetic energy of the moving coolant. The latter may be supplemented by inertia of the pump rotor assembly, if the failure is simply a loss of electric power to the pump motor. In this case, there must be added to the normal pressure drop through the primary loop only that of a windmilling rotor, and this only after it has ceased to contribute any pumping effort from its own inertia. The other possibility is that the pump fails because its rotor assembly is mechanically jammed due to interference or breakage. This is the more pessimistic and therefore the limiting case for design, since the pump can contribute no kinetic energy and offers a much higher resistance to flow.

Equations have been derived for predicting flow coastdown rate with a freely rotating impeller, but in application they break down because they require a knowledge of the coastdown rate of pump rotation, and of its efficiency over the range of flow/rpm ratios experienced during the coastdown. These are both indeterminate for the present application, but it can safely be stated the transient values of pumping efficiency will drop to a very low level, and the internal hydraulic friction of a shaft seal pump would be a major factor in using up its own kinetic energy during coastdown. Therefore, the overall contribution of pump rotor kinetic energy will be small, and the two cases will differ essentially by the difference in negative head of the pump.

Because it is by far the more conservative case, only the case of the frozen impeller will be examined. In selecting from among the equations which have been derived for predicting flow coastdown rate, the most conservative one has been selected, with the added justification that it best fits the data from the only indicative test on an SM type plant.

5.5.2 Basic Formulae

5.5.2.1 Nomenclature

A	Pipe cross sectional area or heat transfer area	ft^2
a	Constant defined by eq. 1	sec.
b	Constant defined by eq. 2	sec^{-1}
g	Acceleration due to gravity	ft/sec^2
h	Head loss across inoperative pump	ft
H	Loop head at normal flow rate	ft
H_0	Total loop head at normal flow rate including h	ft

L	Length of pipe	ft
F_o	Coolant flow under normal operating conditions	ft^3/sec
t	Time after pump failure occurs	sec
R/R_o	Ratio of coolant flow at time t to flow under normal operating conditions	ratio
$\sum L/a$	Summation of pipe "length over area"	ft^{-1}

5.5.2.2 Equations Used

In calculating the rate of pump coastdown after an accident, the rate of loss of kinetic energy from the moving coolant is equated to the rate of power consumption by friction of the loop. Based on continuity of instantaneous flow rates throughout the loop, and overall pressure drop proportional to the 1.8 power of the flow rate, a differential equation is derived with which the normalized flow rate can be integrated over the range of flow, yielding:

$$\left[\frac{R}{R_o}\right]^{0.8} = \frac{a}{t + a}$$

where

$$a = \frac{\sum_{i=1}^{i} \frac{L_i}{A_i} F_o}{0.8\, g\, h} \qquad (1)$$

For convenience this is rearranged thus:

$$\frac{R}{R_o} = \left[\frac{1}{1 + bt}\right]^{1.25}$$

where

$$b = \frac{1}{a} \qquad (2)$$

The experimental data from the SM-1 supports an exponent of 1.10 to 1.20 for the above equation and the reference test did not have a stuck pump impeller. The latter would push the exponent down closer to unity, but this is in the direction away from a conservative assumption.

5.5.2.3 Application to SM-2

Constants used in the SM-2 analysis are:

H = 113.9 ft

F_o = 7800 gpm = 17.378 ft^3/sec

h (frozen impeller) = 100 ft

$\sum \frac{L}{A}$ = 53.25 ft^{-1} (see section 3.4.3.2)

(H_c) frozen impeller = 213.9 ft

5.5.2.4 $\sum \frac{L}{A}$, SM-2 Primary Loop

Reactor Vessel

A. Inlet section

L = 1.688 ft

A = 5.95 ft^2

$$\frac{L}{A} = \frac{1.688}{5.95} = 0.284\ ft^{-1}$$

B. Return section

Section I: L = 0.354 ft

A = 10.10 ft^2

$$\frac{L}{A} = \frac{0.354}{10.10} = 0.031\ ft^{-1}$$

Section II: L = 1.979 ft

A = 4.15 ft^2

$$\frac{L}{A} = \frac{1.979}{4.15} = 0.477\ ft^{-1}$$

C. Outlet sections

Section I: L = 3.96 ft

A = 5.59

$$\frac{L}{A} = 0.709\ ft^{-1}$$

Section II: L = 0.917 ft

A = 4.50 ft^2

$$\frac{L}{A} = 0.203\ ft^{-1}$$

Piping

A. Cold Leg, Generator to Pump

$$\frac{L}{A} = \frac{6.12 \text{ ft}}{0.9397 \text{ ft}^2} = 6.52 \text{ ft}^{-1}$$

B. Pump to Reactor

$$\frac{L}{A} = \frac{7.41 \text{ ft}}{0.7213 \text{ft}^2} = 10.28 \text{ ft}^{-1}$$

C. Reactor to Generator

$$\frac{L}{A} = \frac{10.75 \text{ ft}}{0.7213 \text{ ft}^2} = 14.9 \text{ ft}^{-1}$$

D. Steam Generator Inlet and Outlet

$$A = 1/2\left[\pi D^2/4\right] = 5.03 \text{ ft}^2 \quad \frac{L}{A} = \frac{2.70}{5.03} = 0.528 \text{ ft}^{-1}$$

$$L \text{ equiv} = 2.70 \text{ ft}$$

E. Steam Generator Tubular Volume

$$A = N_t \times A_{int} = 681 \times 0.00210 = 1.43 \text{ ft}^2$$

$$L = 22.3 \text{ ft} \; \frac{L}{A} = \frac{22.3}{1.43} = 15.6 \text{ ft}^{-1}$$

Core

A. 1st Pass

$$L = 1.831 = \frac{L}{A} = \frac{1.831}{0.9249} = 1.982 \text{ ft}^{-1}$$

$$A = 0.9249$$

B. 2nd Pass

$$L = 1.831 \frac{L}{A} = \frac{1.831}{1.0652} = 1.721 \text{ ft}^{-1}$$

$$A = 53.25 \text{ ft}^{-1}$$

$$\sum \frac{L}{A} = 53.25 \text{ ft}^{-1}$$

5.5.3 Conclusions

Based on the preceding data, the predicted flow rate expression is as follows:

$$b = \frac{0.8 \times 32.2 \times 213.9}{53.25 \times 17.38} = 5.95$$

$$R/R_o = \left[\frac{1}{1 + 5.95}\right]^{1.25}$$

Figure 5-2 presents a plot of R/R_o versus t for the first 6 sec after pump failure employing the above relation.

5.6 PRIMARY SYSTEM KINETICS

5.6.1 Pressurizer Design Criteria

The primary function of the pressurizer is to maintain the primary loop at the operating pressure. The other function of the pressurizer is to contain sufficient steam and liquid to absorb the surge of primary liquid volume without excessive pressure fluctuation. These primary liquid volume surges result from changes in the average temperature due to abrupt changes of electrical load of the power plant.

The pressurizer is designed to limit pressure surges to + 150 psi, to comply with the structural design, while encountering an instantaneous load perturbation from full load to 1 percent. It should also limit negative surges to -125 psi to comply with thermal analysis requirements during a load increase from 5 percent to 100 percent in 60 seconds. The loss of load can occur as an instantaneous step function whereas the load gain can only occur as a ramp function.

Since these are the most drastic load changes, intermediate changes such as 100-75 percent, 75-100 percent or 5-30 percent load would be of less magnitude and do not affect pressurizer size.

5.7 METHODS, EQUATIONS, AND MODEL

5.7.1 Methods

A complete set of simultaneous differential equations to describe the physical system and determine primary system and component response to changes in steam generator loads were prepared, and then programmed for solution on an analog computer.

The general kinetic model used for the primary system is that developed in APAE No. 38[4] and APAE Memo No. 127[5], and later expanded for 2 pass core in APAE Memo 269[6]. The validity of the basic model in representing plant transients introduced by load perturbations has been demonstrated by comparison with SM-1 plant data.

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The model was placed on the Ease Analog Computer and runs made to determine the pressurizer characteristics dur to changes in load from 100 percent to 1 percent and ramp increase in load of 5 percent to 100 percent in 60 seconds.

5.7.2 Differential Equations

Differential equations of the behavior of the SM-2 primary system were derived in APAE Memo 269[6].

5.7.2.1 Pressurizer Characteristics

Because the total number of amplifiers required to program the primary system with pressurizer on the analog computer was greater than the amplifiers available, the problem was treated as two uncoupled parts. The problem was easily divided into the two components, one to study the behavior of the pressurizer, the other to couple the results of the first, with pressure-volume relationship approximated as a linear function.

With this approximation, the pressurizer circuitry can be simplified and expressed as:

$$p(t) - p_D = S_{pos} \Delta v_{tot}(t), \qquad \Delta v \leq 0$$

$$p(t) - p_D = S_{neg} \Delta v_{tot}(t), \qquad \Delta v \geq 0$$

Slope S is obtained from the study of the pressurizer behavior which accounts for condensation on metal walls and liquid surface.

5.7.3 Analog Circuit Diagram

The circuit diagram for wiring the electronic analog computer is shown in Fig. 5-3.

5.7.3.1 Time and Amplitude Scaling Factors

Time and amplitude scaling factors of unity were used because of the speed of the transients and ease of interpreting computer results. Proper voltage levels were then obtained by the use of multiples or fractions of the variables.

5.7.3.2 Potentiometer Settings

A listing of servo-set potentiometer settings and plant constants is given in APAE 269. Each setting is stated in terms of both plant symbols and specific numerical values.

5.7.3.3 Analog Computer Model Response

The behavior of the integrated reactor and plant kinetic equations have been investigated on the analog computer. Only the response to load perturbations were considered since primary pressure variations due to control rod perturbations would be of less magnitude than those due to the worst load changes considered.

The reaction of the system temperatures and pressure following a step reduction in load from 100 to 1 percent (of 28 mw) is shown in Fig. 5-4. For a pressurizer volume of 52.55 ft^3, the maximum system pressure rise is +150 psi, which occurs at 150 sec after tripping the steam throttle closed.

The maximum expected rate of increase of load is 5 to 100 percent in 60 seconds. Figure 5-5 shows the system response to this load increase. The system essentially reaches equilibrium after 3 min, during which time the system pressure dipped to a minimum of -125 psi.

5.7.4 Pressurizer Sizing

The computer model considers the condensation of vapor on the metal walls and on the liquid interface rather than assuming adiabatic compression of the steam. During the outsurge, flashing is considered to take place within the pressurizer, thus aiding in reducing the size of the negative pressure surge. For further details see APAE Memo 266[1].

It has been established in Fig. 5-4 that during a loss of load from 100 percent to 1 percent the expansion of the primary system coolant causes an insurge of 6.1 ft^3 to enter the pressurizer. Since it is desired to limit the positive pressure surge to + 150 psi, the steam volume is derived using the insurge volumes. Entering Fig. 5-6 at + 150 psi, one obtains the ratio of insurge volume to initial steam volume as 0.159. Thus the initial steam volume is 38.4 ft^3.

The initial steam volume divided by the system volume outsurge of 3.2 ft^3 for a ramp increase in load 5 pervent to 100 percent in 60 seconds gives a ratio of 12.0. Entering Fig. 5-7 with this ratio one gets a ratio of initial water volume to surge volume of 2.0 and an initial water volume of 6.4. The total pressurizer volume required to limit the positive and negative pressure surges within those specified by design is 44.8 ft^3. However, during an outsurge, there is not enough liquid to adaquently protect the heaters from damage, so an additional 7.7 ft^3 of water was added to establish a condition of having at least 3 in. of water above the heaters at all times. Thus, the total SM-2 pressurizer volume is 52.55 ft^3.

5.7.5 Conclusion and Recommendations

A pressurizer vessel containing 38.7 ft^3 of steam and 13.85 ft^3 of water was selected as being adequate for any primary system volume changes that would be encountered in the operation of the power plant, and offer complete protection of the heaters at all times.

The primary system is designed under code regulations to structurally withstand an internal pressure of 2200 psia. This design pressure provides for a 150 psi pressure surge from the normal operating pressure of 2000 psia without tripping the high pressure scram. The pressurizer design is, therefore, considered adequate to meet the demands of extreme plant load perturbations.

5.8 DECAY HEAT REMOVAL

5.8.1 Introduction

The analysis of the processes taking place in the SM-2 reactor during decay heating is reported in this section. The primary coolant pump is considered to have failed either because of mechanical malfunction or loss of power. Such a failure would result in a decrease in the primary coolant flow rate and result in a reactor scram initiated by the flow sensing device. The reactor power due to fission and prompt radiation during scram is governed by the fundamental reactor kinetic equations for a system with a large added negative reactivity. A sizeable quantity of energy continues to be released after shutdown because of the decay of fission products in the core. The complete problem, from initial pump failure to a time when natural circulation has been established, can be broken down into two parts because of the nature of the events taking place and the difference in time scale.

Part I

This period begins at the instant of pump failure and extends some 5 or 10 seconds later when natural circulation becomes the dominant factor in determining coolant flow rate. During this period it is necessary to investigate the temperature distribution within the core. The combination of a high heat flux in the core before and during the scram together with the fact that the coolant flow rate is decreasing very rapidly produces a situation where damage to the core from boiling could occur. This part is independent of this primary system analysis and has been investigated under the core design contract and was reported in APAE No. 69.[2]

Part 2

The second period begins when flow due to natural circulation becomes greater than that due to pump coastdown. It continues until all temperatures are decreasing and is referred to as the decay heat removal problem.

The coupled nature of the problem arises describing the following simultaneous processes.

A. Primary coolant mass flow rate decreases as its kinetic energy is converted to thermal energy by mechanical friction through the pipe loop. The decreased fluid velocity results in increased piping lags through the loop and decreased heat transfer characteristics of both the core and the steam generator. Eventually flow is established by natural circulation dependent on the physical configuration of the system and differences in fluid densities at specific points.

B. The core outlet temperature tends to decrease because of the reduction in fission power output, the decreased heat transfer characteristics of the core and the release of energy due to fission products will to some extent cause a temperature increase in the core.

C. Energy is removed from the system through the steam generated in the heat exchanger. The amount of heat that can be removed is dependent upon the load applied to the steam generator and the amount of water in the secondary of the steam generator.

5.8.2 Nomenclature

A	Flow or heat transfer area, ft^2
b	Pump coastdown characteristics
c_p	Specific heat at constant pressure, Btu/lb^{o}F
d	Vertical distances, ft
g	Acceleration due to gravity, ft/sec^2
G	Mass flow rate, lb/hr-ft^2
$H(o)$	Head drop around coolant loop under steady state conditions, ft H_2O
$L(t)$	Load factor of steam generator power output, dimensionless
N_{nu}	Nusselt number
N_{pr}	Prandtl number

5.8.3 Basic Kinetic Equations for Decay Heat Removal Analysis

5.8.3.1 Thermal Kinetics of the Core

The core design for the SM-2 is a two pass system with the flow directed upward in each pass. Within each pass, the fuel plates and the coolant are treated as separate lumped systems.

The following assumptions were made for this analysis of the core (7)(8):

A. The time of coolant transient through the core is small compared to the magnitude of the other delays in the system.

B. The temperature difference across the core can be effectively replaced by the arithmetic mean temperature difference.

C. Fuel plates are considered to be at a mean temperature and the difference between the mean fuel plate temperature and the mean fuel plate temperature and the mean coolant temperature is proportional to the heat transferred. The proportionality factor is the heat transfer coefficient times heat transfer area:

$$P_d = (UA)_o \quad \theta_{FC} \tag{1}$$

A heat balance can be performed on the fuel plate in which the storage rate is equal to the heat generation rate minus the heat transfer rate. This yields for the first pass:

$$W_x C_x \frac{dTx}{dt}(t) = 2\phi_x \quad Q(t) = V_x A_x \left(T_x(t) - T_a(t)\right) \tag{2}$$

The heat transfer coefficient is considered to be a function of the first rate as expressed by the turbulent heat transfer equation of Nusselt:

$$N_{Nu} = \frac{hD}{k} = 0.023 \,(N_{RE})^{0.8} \,(N_{PR})^{0.4}$$

where

$$N_{RE} = \text{Reynolds Number} = \frac{DG}{\mu} \tag{3}$$

N_{PR} = Prandtl Number

N_{Re} Reynold's Number

P Pressure, lbs/ft^2

Q (t) Power output of core Btu/sec

R Flow rate, lbs/sec

t Time, sec

T Temperature, ^{o}F

U Heat transfer coefficient, Btu/sec $-^{o}F-ft^{2}$

V Volume, ft^{3}

W Weight, lb

x linear pipe length, ft

Z Vertical disturbance, ft

α Fraction of power generated in fuel plates and cladding

γ volume coefficient of expansion for primary coolant, $ft^{3}/^{o}F$

ϕ_1 Fraction of core power produced in first pass

ϕ_2 Fraction of core power produced in second pass

θ Temperature difference at design load, ^{o}F

σ Time lag, sec

ρ Density of fluid, lbs/ft^{3}

5.8.3.2 Subscripts

A Mean core coolant condition, first pass

B Mean core coolant condition, second pass

C Mean core coolant condition,

D Design condition during steady state operation at full power

E Mean exchanger tubing condition

F Mean fuel plate condition

F Pipe friction factor

G Mean steam generator condition, primary tube side

i Summation index

o Initial condition

P Resistance due to pump friction

S Main steam conditions in generator corresponding to saturation

x Mean fuel plate condition in the first pass

y Mean fuel plate condition in the second pass

1 to 9 Thermodynamic properties at points shown in the schematic Fig. 5-8.

Thus, the following relationship would hold:

$$U_x A_x = (U_x A_{xb} \left[\frac{R}{R_o}\right]^{0.8} \tag{4}$$

For convenience of solution on the analog, the exponent 0.8 can be replaced by unity. This will be conservative when the ratio (R/R_o) is less than one (i.e. a number less than unity raised to the 0.8 power is greater than the number itself).

The resulting lumped equations for the fuel plates and the average coolant temperature in the first pass become:

$$\frac{dT_x(t)}{dt} = \frac{\alpha \phi_x Q(t)}{W_x C_x} - \frac{P_d}{W_x C_x \theta_{fcx}} \left[\frac{R}{R_o}\right] \left[T_x(t) - T_a(t)\right] \tag{5}$$

$$\frac{dT_a(t)}{dt} = \frac{(1-\alpha)\phi_x Q(t)}{W_a C_a} + \frac{P_d}{W_a C_a \theta_{fcx}} \left[\frac{R}{R_o}\right] \left[T_x(t) - T_a(t)\right] - \frac{R}{W_a}\left[T_2(t) - T_i(t)\right] \tag{6}$$

A further simplification of (6) can be made by assuming that all of the fission product energy is released by the core. It has been proven[8] that the amount of decay heat generated in the coolant channel surrounding a plate is of the order of magnitude of 1.5 percent, therefore α is assumed to be unity. Similar equations were written for the second pass.

5.8.4 Thermal Kinetics of the Steam Generator

The steam generator is considered as a lumped system in in much the same manner as the core. The additional assumption that the thermal capacity of the tubes be lumped with the secondary fluid can be made for simplicity.

Performing a heat balance on the primary and secondary side of the steam generator we have:

$$\frac{d\,T_g\,(t)}{dt} = \frac{C_g\,R}{W_g C_g}\left[T_7(t) - T_8(t)\right] - \frac{(U_g A_g)_o}{W_g C_g}\left[\frac{R}{R_o}\right]\left[T_g(t) - T_s(t)\right] \tag{7}$$

$$\frac{dT_s(t)}{dt} = \frac{(U_g\,A_e)_o}{W_e C_e + W_s C_s}\left[\frac{R}{R_o}\right]\left[T_G\,(t) - T_s(t)\right] - \frac{Q_d L(t)}{W_E C_E + W_s C_s} - \frac{Q_L(+)}{W_E C_E + W_s C_s} \tag{8}$$

The last two terms on the right of (8) are the energy used to supply steam to the secondary and to heat makeup feedwater. These terms are generally combined to form the heat sink applied to the secondary.

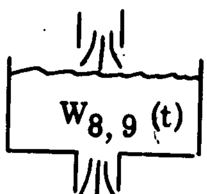
5. 8. 5 Kinetics of Plenum Chambers and Piping

There are four plenum or mixing chambers in the SM-2 primary system (Fig. 5-8) at the core inlet and outlet and the steam generator inlet and outlet. The core inlet plenum chamber is comparatively small and can be considered as an extension of the cold leg piping. (5)

With the assumption that complete mixing occurs in the chamber, an energy balance yields the following:

R_8 (t)

T_8 (t)



R_9 (t)

R_9 (t)

$$C\left[R_8\,(t)\,T_8\,(t) - R_g(t)\,T_g(t)\right] = \frac{C_d\left[W_{8,9}(t)\,T_9(t)\right]}{dt} \tag{9}$$

where $R_8(t) = R_g\,(t) = R$

$W_{8,9}\,(t) = W_{8,9}$

then

$$R\,T_8(t) - R\,T_9\,(t) = W_{8,9}\,\frac{dT_9\,(t)}{dt} \tag{10}$$

The fluid flow velocity profile in the primary piping exhibits a profile that is essentially flat and the mathematical relationship of the transport leg can be expressed in terms of slug flow.

$$T_1\,(t + \tau_{9,1}) = T_9\,(t) \qquad \tau_{9,1} = W_{9,1}/R$$

$$T_3\,(t + \tau_{2,3}) = T_3\,(t) \qquad \tau_{2,3} = W_{2,3}/R$$

$$T_6\,(t + \tau_{5,6}) = T_5\,(t) \qquad \tau_{5,6} = W_{5,6}/R \tag{11}$$

By combining the first order lag of the mixing chamber and the transport lag of connected piping with the equation defining arithmetic mean temperature one arrives at:

$$\frac{dT_1\,(t + \tau_{9,1})}{dt} = \frac{2}{\tau_{8,9}}\;T_g\,(t) - \frac{1}{\tau_{8,9}}\;T_7\,(t) - \frac{1}{\tau_{8,9}}\;T_1\,(t + \tau_{9,1}) \tag{12}$$

$$T_2\,(t) = T_3\,(t + \tau_{2,3}) = 2T_a\,(t) - T_1\,(t) \tag{13}$$

$$\frac{dT_6\,(t + \tau_{5,6})}{dt} = \frac{2}{\tau_{4,5}}\,T_b\,(t) - \frac{1}{\tau_{4,5}}\;T_3(t) \quad \frac{1}{\tau_{4,5}}\,T_6(t + \tau_{5,6}) \tag{14}$$

The piping delays were represented on the analog computer by means of second order Pade delay circuits. (8)

5.8.6 Coolant Flow Analysis

The rate of change of momentum of an element of fluid is equated to the sum of the forces acting on it. These forces are gravity, friction and fluid inertia. These forces all tend to resist the flow, and correspond to pressures that must be balanced by the pump during steady state operation. (8)

$$(\Delta P_p) = \sum_i \left[\frac{\Delta X_i}{A_i g} \quad \frac{d\,R_i}{dt} = (\Delta P_f)_i + \rho_i\,\Delta Z_i \right] \tag{15}$$

In order to apply this to the reactor model, the following assumptions are made:

The coolant is an incompressible fluid.

The increments of pipe length are sufficiently short so that the integral around the loop may be replaced by a sum over the element of length

The density of the fluid at a specific temperature can be measured relative to a reference density ($_o$) by the following relation involving the volume expansion coefficient of the fluid.

$$\rho_i (t) = \rho_o - \gamma T_i \tag{16}$$

The temperature at any instant between two adjacent points defined in Fig. 5-8 varies linearly with position.

Transposing (16) to solve for change in flow rate:

$$\sum_i \frac{\Delta X_i}{A_i g} \frac{d R_i}{dt} = \sum_i (\Delta P_p) - (\Delta P_f)_i - \rho_i \Delta Z_i \tag{17}$$

Even though this equation is for steady state, it is applicable after pump failure. However, for the case of a frozen impeller, the pressure resistance of the pump is a negative quantity.

To evaluate the static pressure term, a simplified view of the system was used (Fig. 5-9) with the convention that flow in an upward direction was positive. Starting at the reactor inlet and performing a sum over the elements of piping calculation, one obtains,

$$\begin{aligned} \sum_i \rho_i \Delta Z_i = & \frac{\rho_1 + \rho_2}{2} L_R - \frac{\rho_2 + \rho_3}{2} L_R + \frac{\rho_3 + \rho_4}{2} L_R \\ & - \frac{\rho_5 + \rho_6}{2} d_2 + \frac{\rho_6 + \rho_7}{2} d_3 + \frac{\rho_7 + \rho_G}{2} L_G \\ & - \frac{\rho_G + \rho_8}{2} L_G - \frac{\rho_8 + \rho_9}{2} d_3 + \frac{\rho_9 + \rho_1}{2} d_1 \end{aligned} \tag{18}$$

Substituting (16) into (18), the following relation is obtained:

$$- \sum_i \rho_i \Delta z_i = \gamma \left[\left(\frac{L_R + d_1}{2}\right) T_1 (t) - \left(\frac{d_2 - L_R}{2}\right) T_5(t) + \left(\frac{d_3 - d_2}{2}\right) T_6(t) \right.$$

$$\left. + \left(\frac{L_G + d_3}{2}\right) T_7(t) - \left(\frac{d_3 - L_G}{2}\right) T_8 (t) + \left(\frac{d_1 - d_3}{2}\right) T_g(t) \right] \qquad (19)$$

The term involving friction pressure drop in (17) may be written as:

$$(\Delta P_f)_i = (\Delta P_p)_o \left[\frac{R}{R_o}\right]^2 \qquad (20)$$

The exponent 2 stems from basic turbulent flow relations for pressure drops in piping. With this substitution, (17) becomes:

$$\frac{dR}{dt} = \frac{- (\Delta P_f + \Delta P_p)_o \left[\frac{R}{R}\right]_o^2 - \sum_i \rho_i \Delta z_i}{\sum_i \frac{L_i}{A_i g}} \qquad (21)$$

5.9 ANALOG MODEL FOR DECAY HEAT REMOVAL

5.9.1 Analog Circuit

The analog circuit diagram used in this analysis is shown in Fig. 5-10. A time scale of one computer second equal to three real seconds has been chosen for this simulation. This was a compromise between the short response time required by the electronic multipliers to avoid drifting and the long response time of the delay circuits to avoid instability. The response of the delay circuits was empirically verified subject to the perturbations imposed by this problem.

An amplitude scaling factor of one volt equals one physical quantity was chosen. Multiples of quantities were generated to keep voltages within the $\pm$ 100 volt limit imposed by the analog computer.

5.9.2 Plant Constants

The plant constants calculated from final design drawings are given below.

C_c	=	1.175	Btu/lb ^oF
C_E	=	0.13	Btu/lb ^oF
C_F	=	0.1185	Btu/lb ^oF
C_L	=	1.185	Btu/lb ^oF
	=	2.0×10^{-5}	sec
P_D	=	2000	psi
Q_D	=	26.538	Btu/sec
R_D	=	852.1	lbs/sec
T_1	=	500.5^o	^oF
T_{1-4}	=	512.3^o	^oF
T_{SD}	=	462.8^o	^oF
T_4	=	525.0^o	^oF
V_a	=	1.6957	ft^3
V_b	=	2.1364	ft^3
V_G	=	31.90	ft^3
V_L	=	110.9	ft^3
$V_{2,3}$	=	11.776	ft^3
$V_{4,5}$	=	26232	ft^3
$V_{5,6}$	=	7.754	ft^3
$V_{6,7}$	=	13.69	ft^3
$V_{8,9}$	=	13.61	ft^3
W_a	=	83.74	lbs
W_b	=	104.0	lbs
W_E	=	6848	lbs
W_x	=	378.3	lbs
W_y	=	331.1	lbs
W_G	=	1565	lbs
W_L	=	5437	lbs
$W_{2,3}$	=	582.9	lbs
$W_{4,5}$	=	1264	lbs
$W_{5,6}$	=	373.7	lbs
$W_{6,7}$	=	659.8	lbs
$W_{8,9}$	=	675.4	lbs
$W_{9,1}$	=	1048	lbs
$\theta FC1$	=	17.10	^oF
$\theta FC2$	=	32.30	^oF
θ GS	=	49.2	^oF
R	=	11.76	lbs/sec
R_o	=	20.00	lbs/sec
$(\Delta P_f)_o$	=	13.55	lbs/ft^2
$(U_x A_x)_o$	=	31.30	Btu/sec-ft^2
$(U_y A_y)_o$	=	24.35	Btu/sec-ft^2

$V_{9,1}$	=	21.13	ft^3	$(U_G A_G)_o$	=	26.54 Btu/sec-ft^2
H_e	=	113.9	ft	ϕ_1	=	0.405
H'_o	=	224.9	ft	ϕ_2	=	0.595
$\frac{\sum L_i}{A_i}$	=	53.25	$\frac{ft}{ft^2}$			

5.9.3 Potentiometer Settings

The values of potentiometer settings used in this simulation are given below. Both the actual magnitude and the physical quantities are given.

1	2/10	0.2000	21	I. C. $T_B(o)$	0.2816
2	$12(3R/W_{2,3})^2$	0.0440	22	$36(R/W_{9,1})$	0.4040
3	$12(3R/W_{2,3})^2$	0.0440	25	2/10	0.2000
4	$36(R/W_{2,3})$	0.7261	26	2/10	0.2000
5	$18(R/W_{2,3})$	0.3631	29	$3\ (R/W_{6,7})$	0.0535
6	$18(R/W_{2,3})$	0.3631	30	Time to Function on Gen.	0.0030
8	$6(\Delta P_f)o/N\ R_o$	0.2464	31	5/10	0.5000
9	$16(d_3-d_1)/N\ R_o$	0.0607	32	$18(R/W_{9,1})$	0.2020
10	$15(d_3-d_2)/N\ R_o$	0.0139	33	I. C. Amp 28	0.1570
12	$15(d_2-L_g)/N\ R_o$	0.0192	35	$3\ (R/W_{6,7})$	0.0535
13	$84\ 2/W_y C_y$	0.1231	37	I. C. T_7 (o)	0.2590
14	$6\ (U_v A_y)./WyCy$	0.3724	38	I. C. T_x (o)	0.3230
15	I. C. $T_y(o)$	0.6046	39	$6\ (U_x A_x)_o/W_x C_x$	0.4225
17	$15(L_G + d_3)/N\ R_o$	0.3346	40	$84\phi_1/Wx\ Cx$	0.0765
18	$15(L_G + d_3)/N\ R_o$	0.3346	41	$6\ (U_x A_x)_o/W_A C_A$	0.1909
19	$6\ (UyAy)o/W_B C_B$	0.1196	42	$6\ R_o/W_a$	0.5277
20	$6\ R_o/W_B$	0.1154	43	I. C. Ta (o)	0.1521

44	I. C. T_g (o)	0.1295	58	$6\,(U_GA_G)o/W_GC_G$	0.0859
46	$15(L_r + d_1)/N\,R_o$	0.0659	59	$6\,(U_GA_G)o/(W_EC_E\text{-}W_SC_S)$	0.0217
47	I. C. $10\,(R/R_o)$	0.0588	60	$3Q_o/(W_EC_E/W_SC_S)$	0.0011
49	$6\,(R/W_{4-5})$	0.0558	61	$36(R/W_{5-6})$	0.1133
50	$3\,(R/W_{4-5})$	0.0279	62	I. C. T_s(o)	0.1295
51	$3\,(R/W_{4-5})$	0.0279	63	$6\,(R/W_{8-9})$	0.1045
52	I. C. T_6(o)	0.2590	64	$3\,(R/W_{8-9})$	0.0522
53	$12(3R/W_{5-6})^2$	0.0107	65	$3\,(R/W_{8-9})$	0.0522
54	$12(3R/W_{5-6})^2$	0.0107	67	$12(3R/W_{9-1})^2$	0.0125
55	$18(R/W_{5-6})$	0.5664	68	$12(SR/W_{9-1})^2$	0.0125
56	$18(R/W_{5-6})$	0.5664	69	$18(R/W_{9-1})$	0.2020
57	$6\,R_o/W$	0.0767	70	1/20	0.0500

5.9.4 Initial Condition for the Analog Simulation

It has already been established[7] that the most serious loss of flow accident is the frozen pump. Therefore all potentiometer calculations have been made for the condition of frozen rotor. It is conceivable that in the case of such an accident the initial mean coolant temperatures for the decay heat problem might be higher than their steady state values, so computer solutions were obtained assuming a 10^oF rise in steady state temperatures. The time to start the problem was taken as 5 sec after scram.

The secondary of the SM-2 will provide at least a rate of power absorption of 3 percent for 100 min (or 5 percent for 60 min) and with the boiler feed pump operative a rate of power absorption of 5 percent for an indefinite period time. The simulation has been investigated applying both 3 percent and 5 percent to the secondary of the steam generator.

5.9.5 Energy from the Decay of Fission Products

The source of heat to the reactor during this phase after pump failure is from the decay of fission products. This quantity has been evaluated in the form of Wigner's formula with constants fitted from recent data;(8)(10)(11)

$$\frac{Q(t)}{Q_o} = 0.05225\, t^{-0.2} - 0.05225\, (t + t')^{-0.2}$$

where

t = time after reactor shutdown (sec)

t' = total time of core operation (sec)

Q_o = reactor initial power (Btu/sec)

Q(t) = reactor power at time t seconds after shutdown (Btu/sec)

In adapting this heat input term for analog computation, the function was calculated for t' equal to 400 days and set into the function generator.

5.9.6 Results

The transient response of the kinetic model was determined for the condition of frozen or locked pump impeller assuming the secondary of the steam generator is a heat sink capable of removing 3 and 5 percent of total load. Analog recordings are shown in Figs. 5-11 (3 percent sink) and 5-12 (5 percent sink).

The analog simulation shows that with 3 percent of full power being removed by the secondary system the temperatures around the loop are oscillating about the steady state values. At no time during the first 12 minutes does any temperature rise more than 35^{o} above steady state conditions.

If 5 percent of full power is removed by the secondary system the temperatures are decreasing 7 to 8 min after pump failure. Again the temperatures have only risen a maximum of 35^{o} above the steady state conditions.

5.9.7 Conclusions

When 5 percent load is absorbed by the secondary, the primary system temperatures have risen to a maximum of 35^{o}F above steady state and are decreasing 8 minutes after scram. At least 5 percent power will be absorbed by the auxiliary equipment indefinitely. Therefore, it is concluded that the SM-2 plant can satisfactorily dissipate the decay heat following a loss of flow accident.

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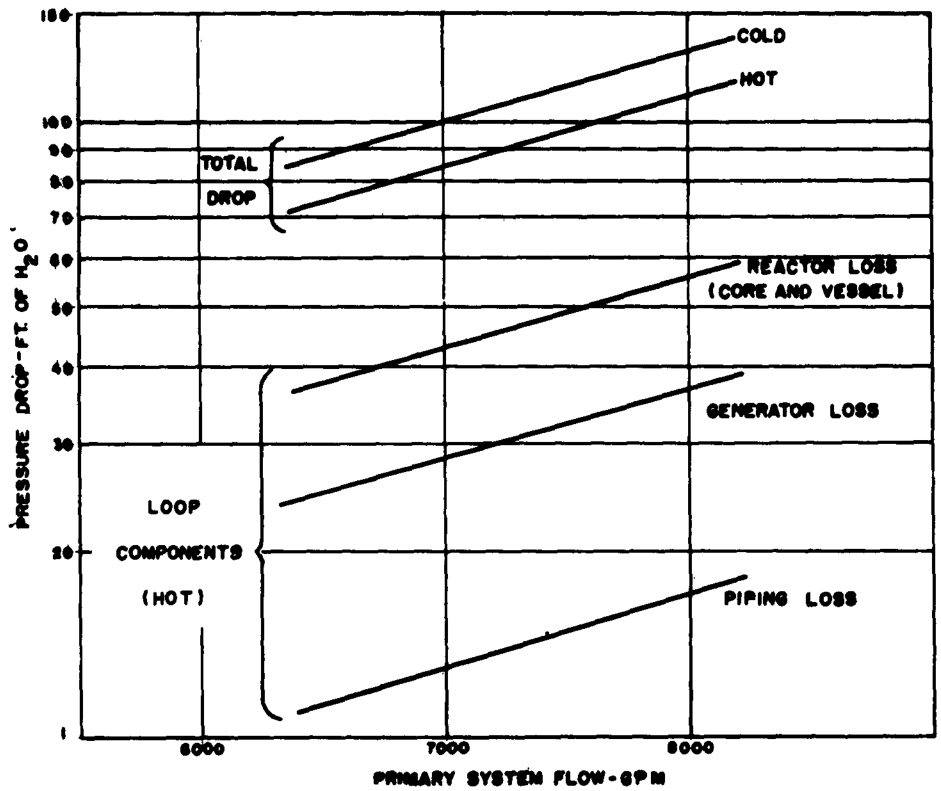
FIG. 5-1

PRESSURE DROP LOSSES VS.
TOTAL FLOW FOR SM-2 PRIMARY
LOOP COMPONENTS



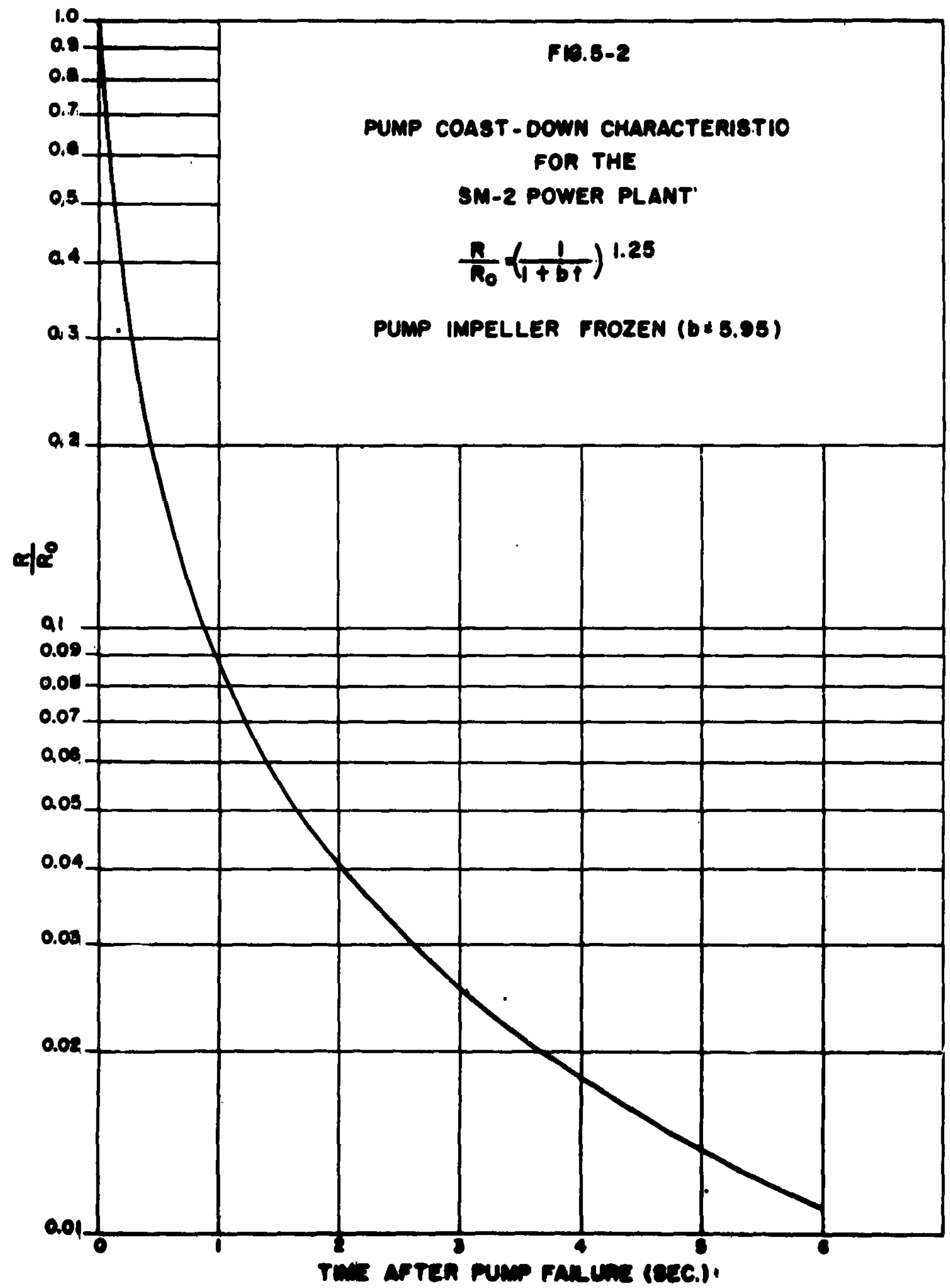
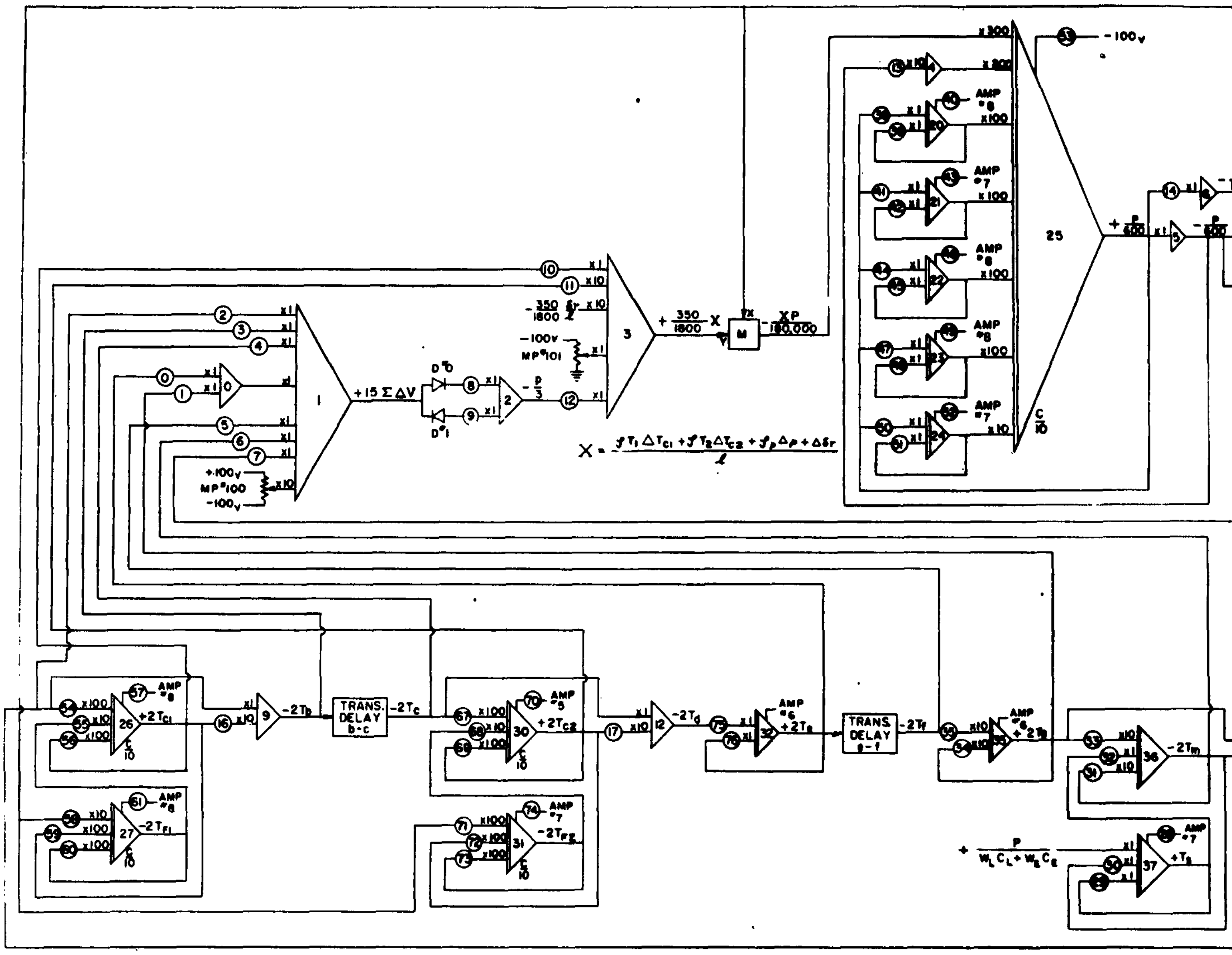


FIG. 5-2
PUMP COAST-DOWN CHARACTERISTIC
FOR THE
SM-2 POWER PLANT
$\frac{R}{R_0} = \left(\frac{1}{1+bt}\right)^{1.25}$
PUMP IMPELLER FROZEN (b = 5.95)
R/R_0
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
1
2
3
4
5
6
TIME AFTER PUMP FAILURE (SEC.)

1



-100v
25
AMP #8
AMP #7
+15 ΣΔV
MP#100
MP#101
+100v
-100v
TRANS. DELAY b-c
TRANS. DELAY e-f
-2Tc
-2Tf
+2Tc1
-2Tf1
+2Tc2
-2Tf2
-2Td
+2Te
-2Tb
-2Tm
+Ts

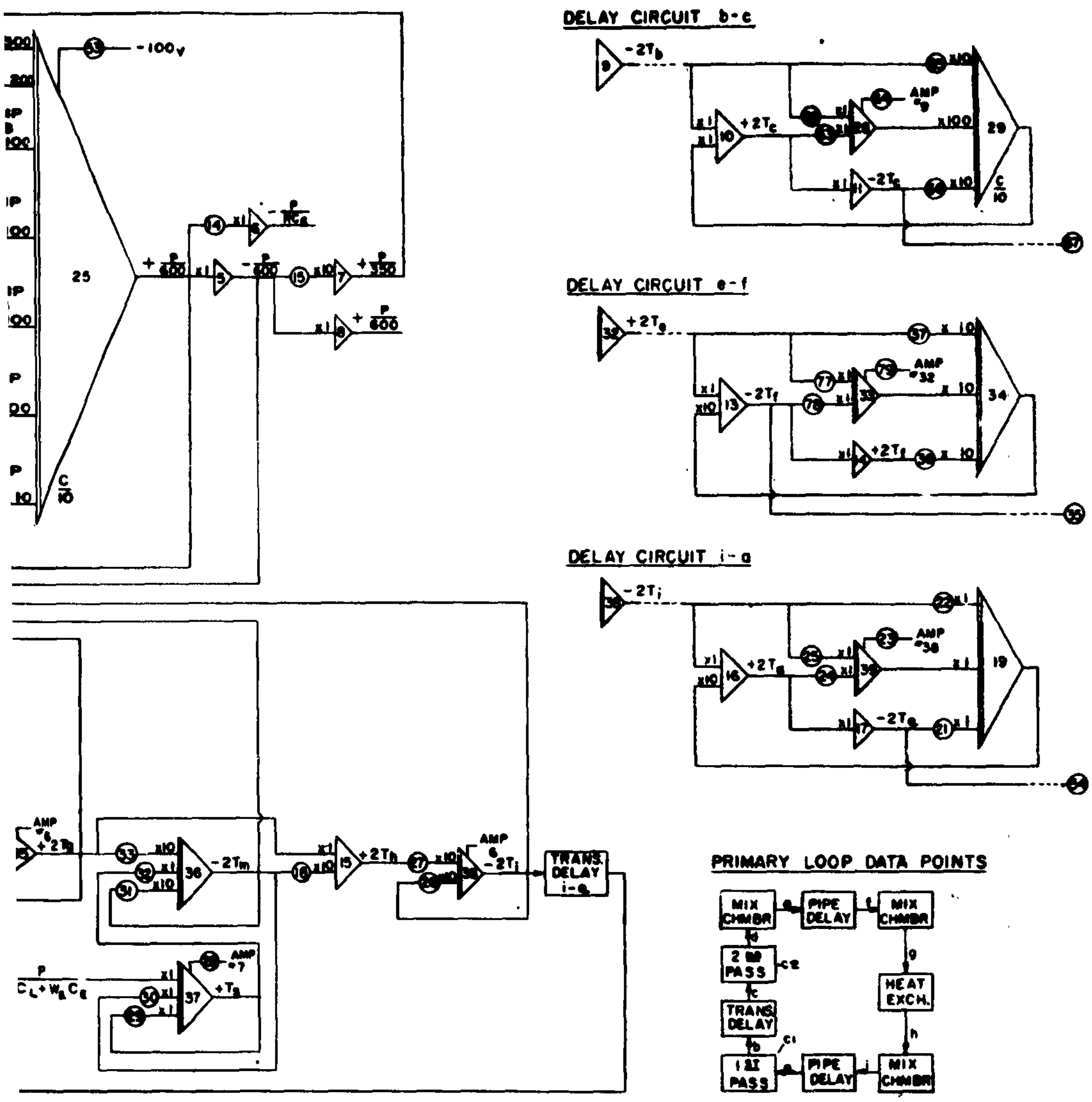
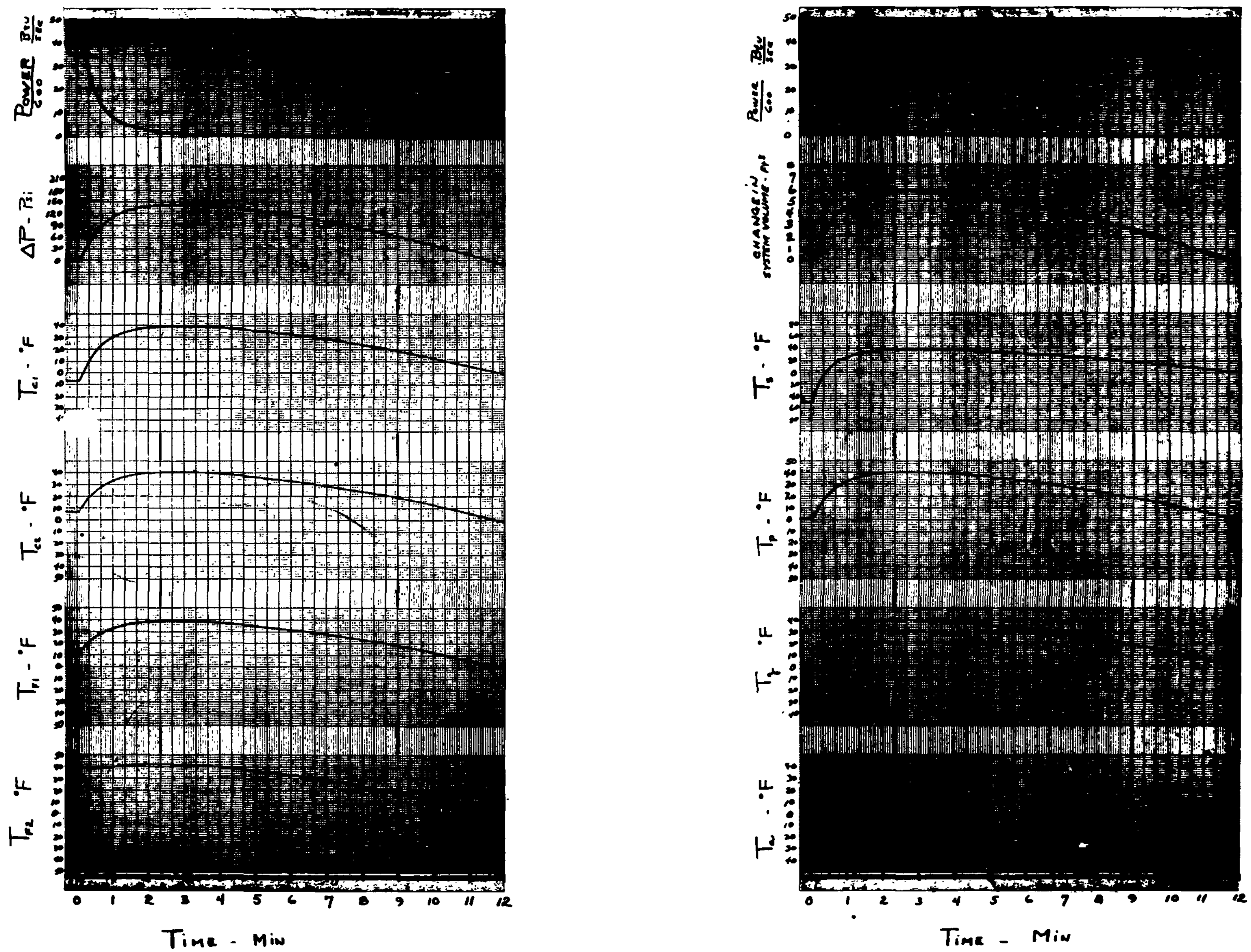
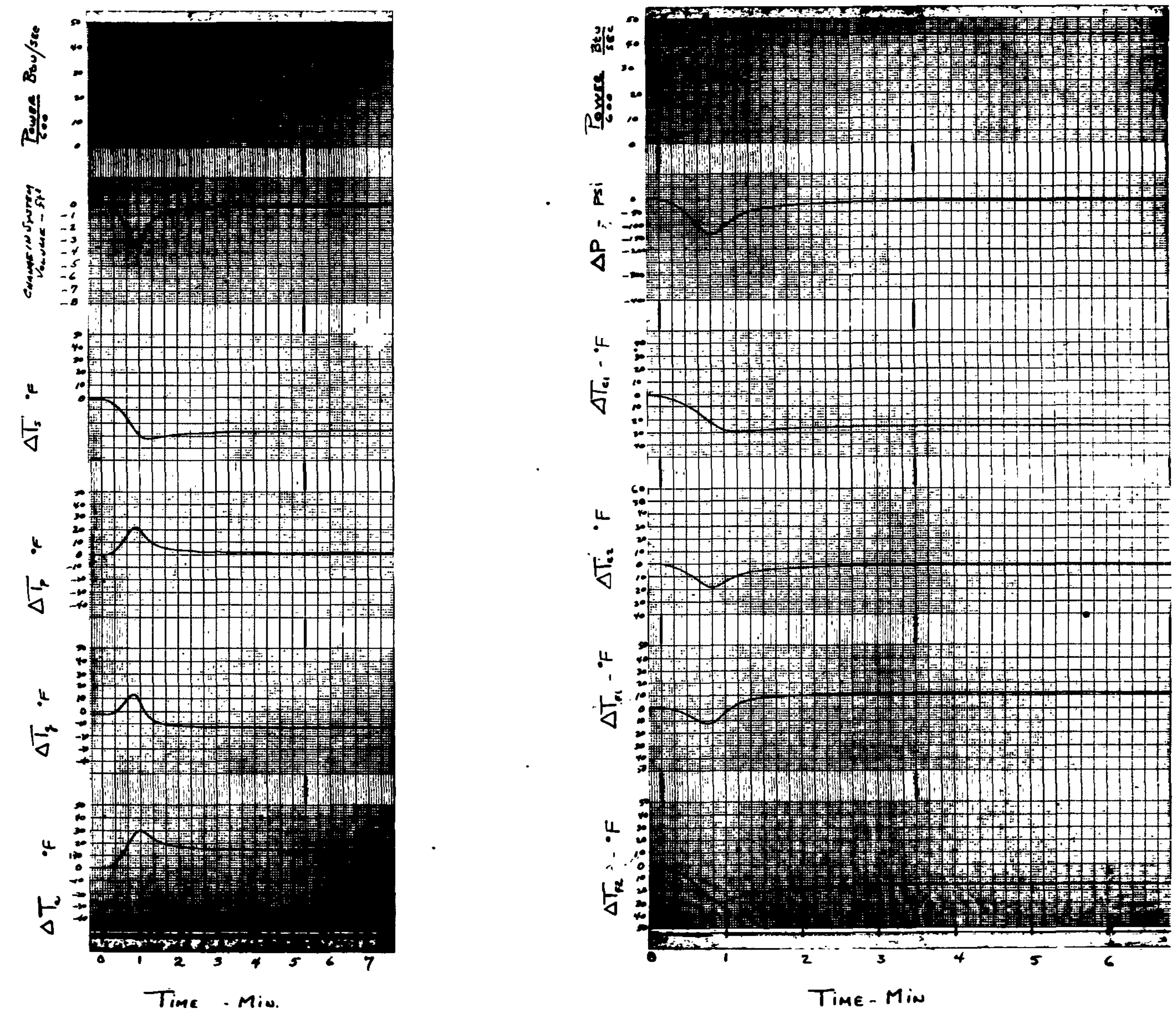


Fig. 5-3 **Circuit Wiring Diagram - Analog Computer - SM-2 Plant Kinetics**



Plant Response to Instantaneous Load Reduction 100% to 1%

Figure 5-4



PLANT RESPONSE TO LOAD INCREASE
OF 5% TO 100% IN 60 SECONDS

Figure 5-5

PRESSURE SURGE VS RATIO OF INSURGE
VOLUME TO INITIAL STEAM VOLUME

Pressurizer Configuration - vertically mounted cylinder with hemispherical heads top and bottom - WITHOUT STANDPIPE

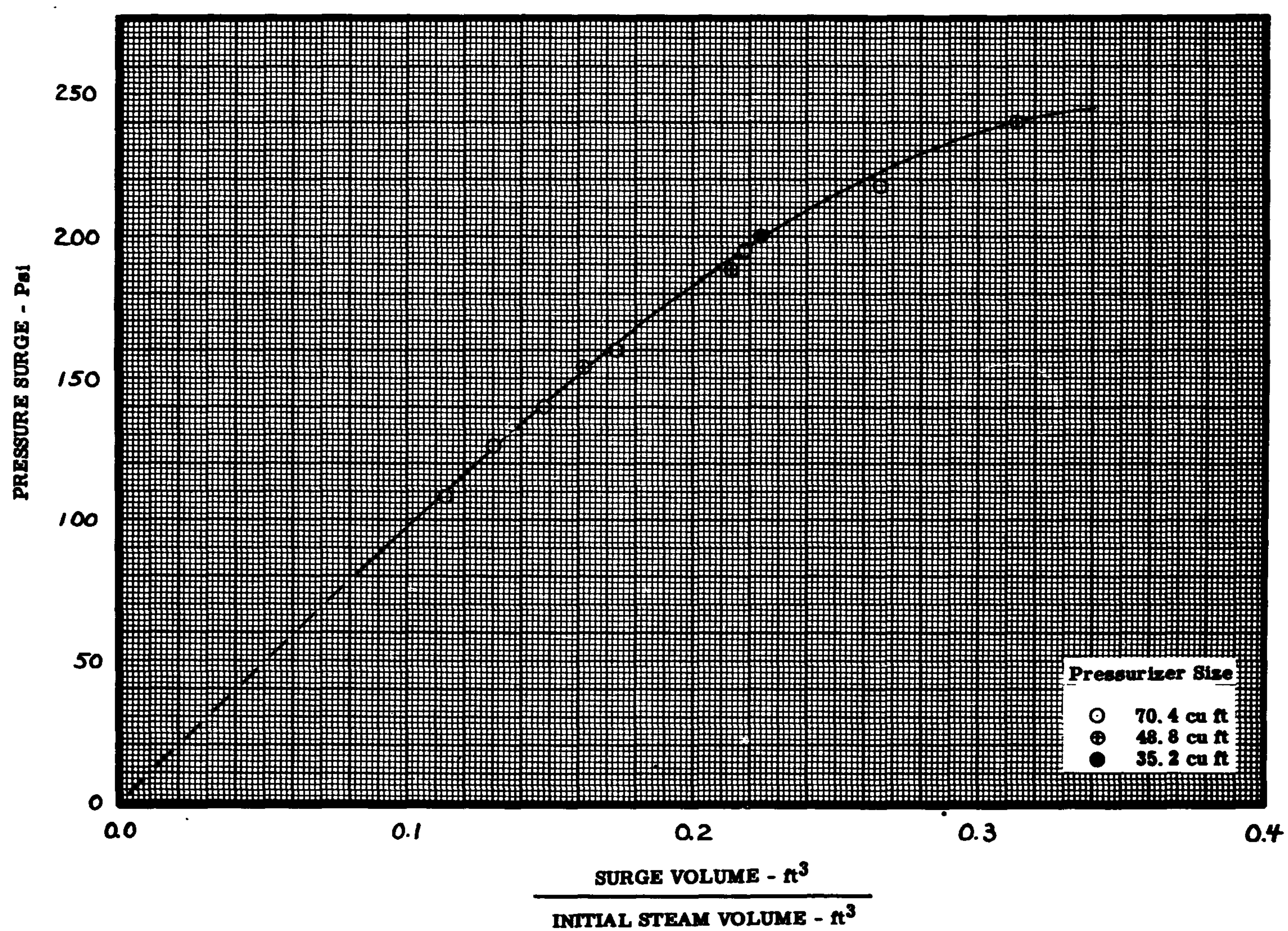
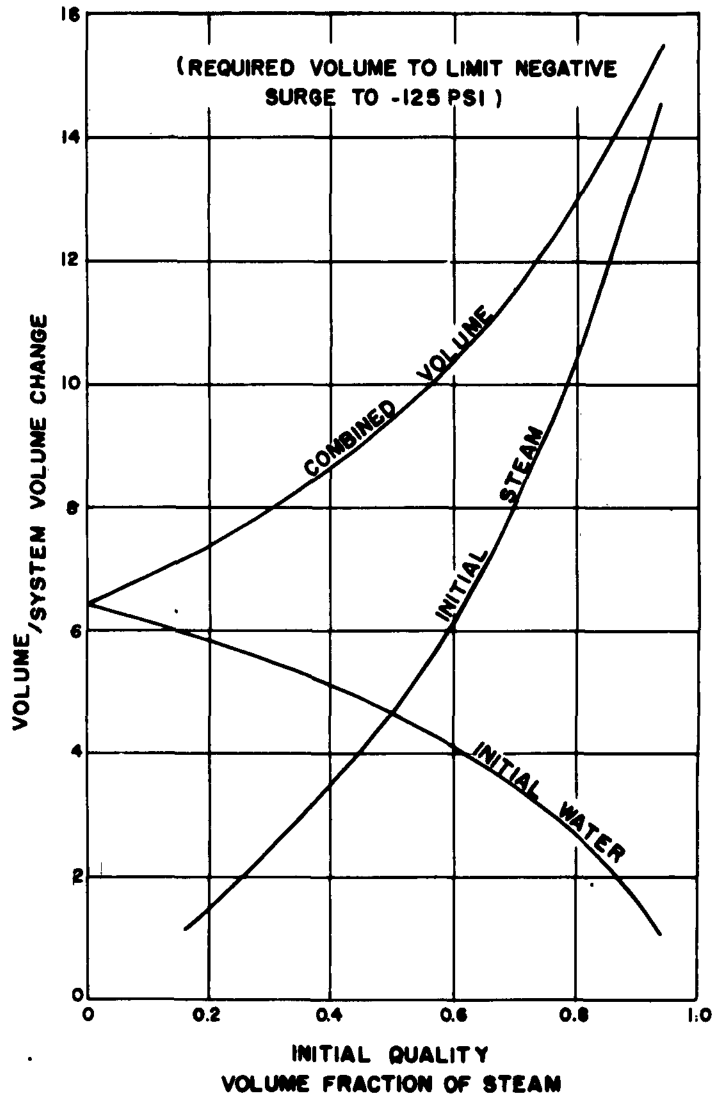
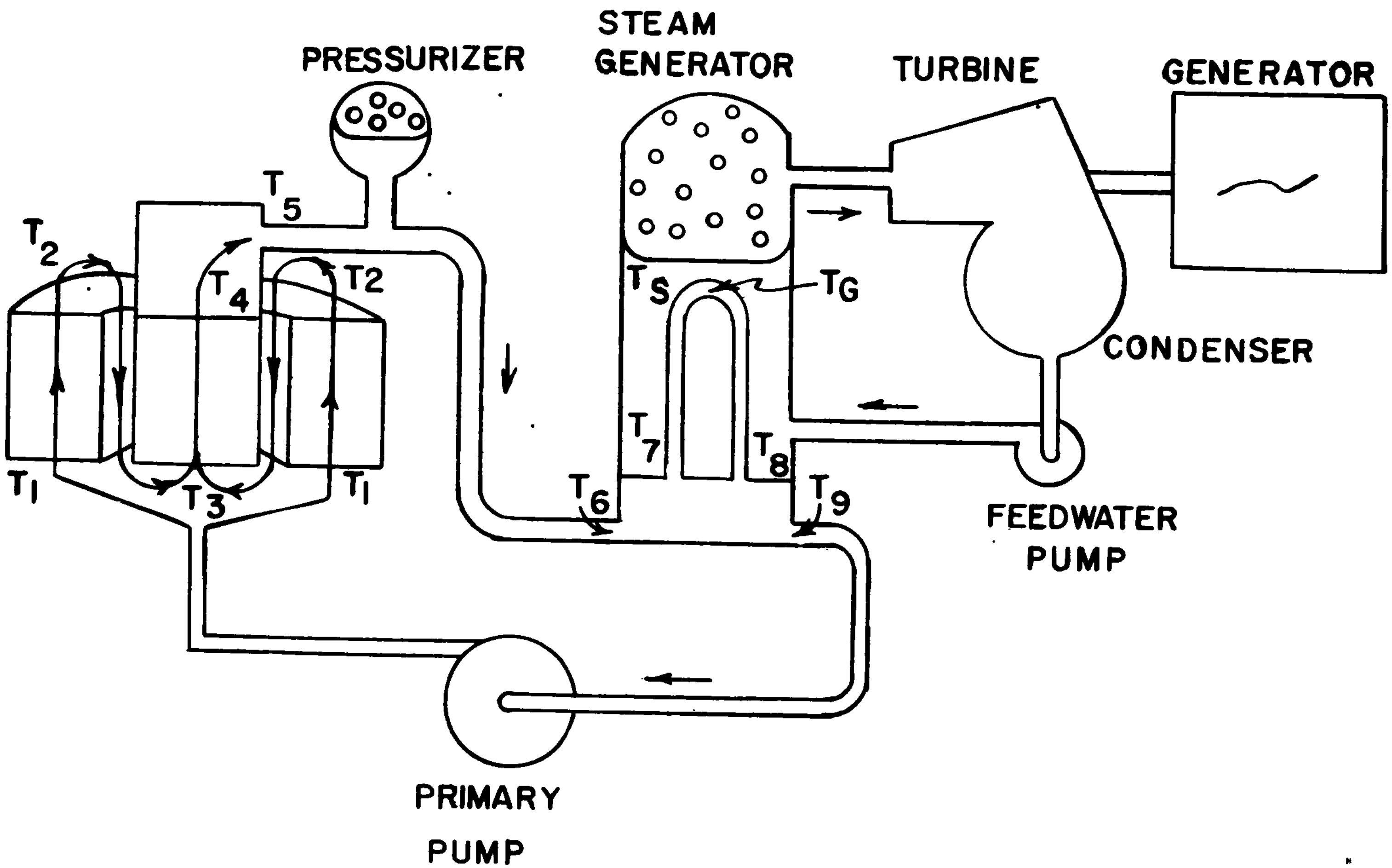


Figure 5-6

FIG. 5-7

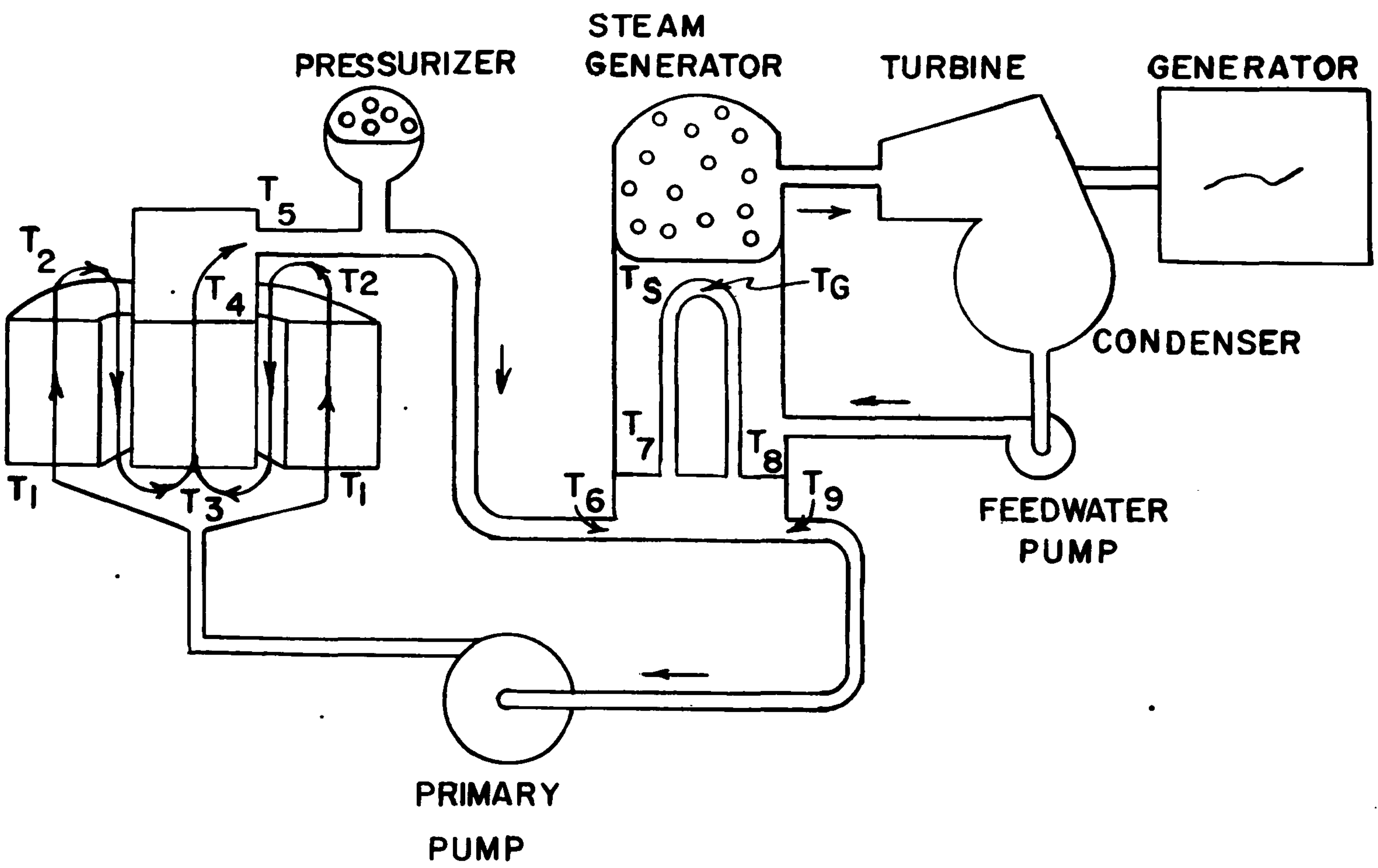
VOLUME OF PRESSURIZER STEAM AND WATER ASSUMING THERMAL EQUILIBRIUM OF WATER AND STEAM @ 2000 PSIA





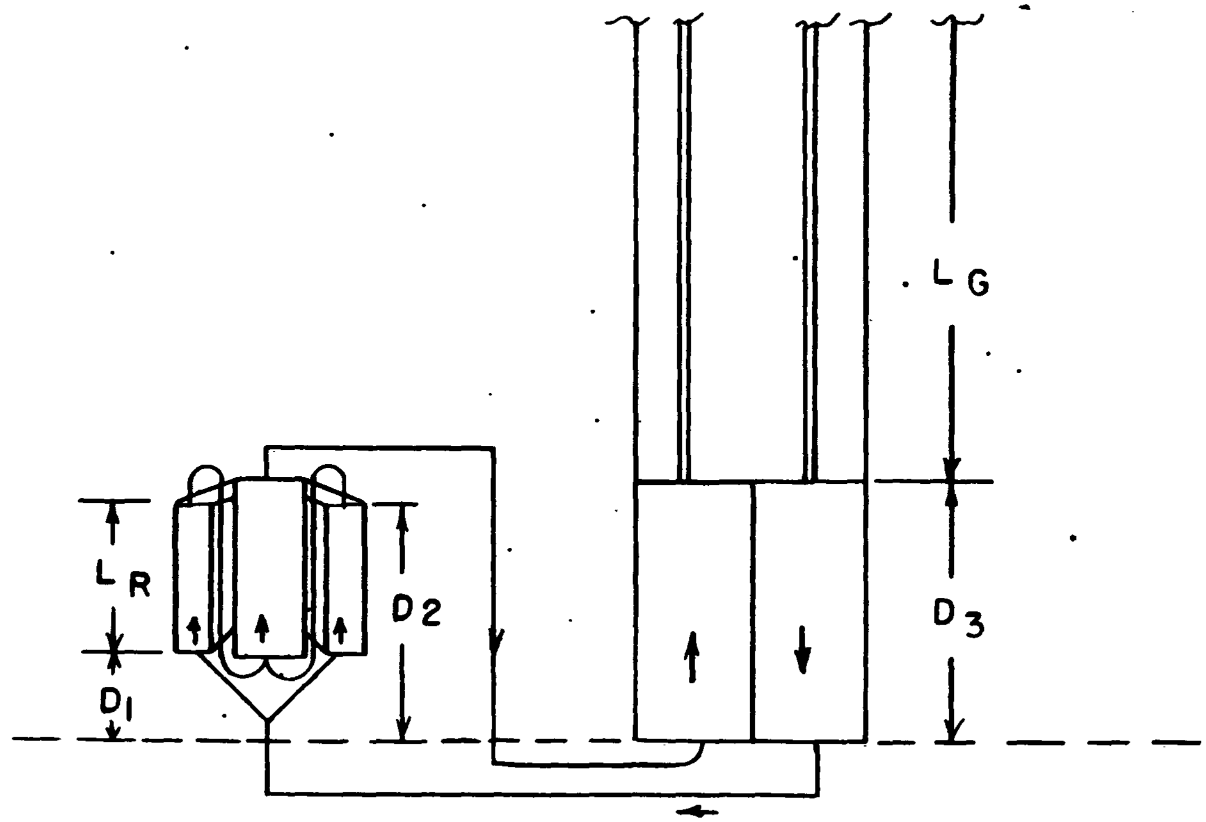
Schematic Diagram Of Sm-2

Figure 5-8



Schematic Diagram Of Sm-2

Figure 5-8

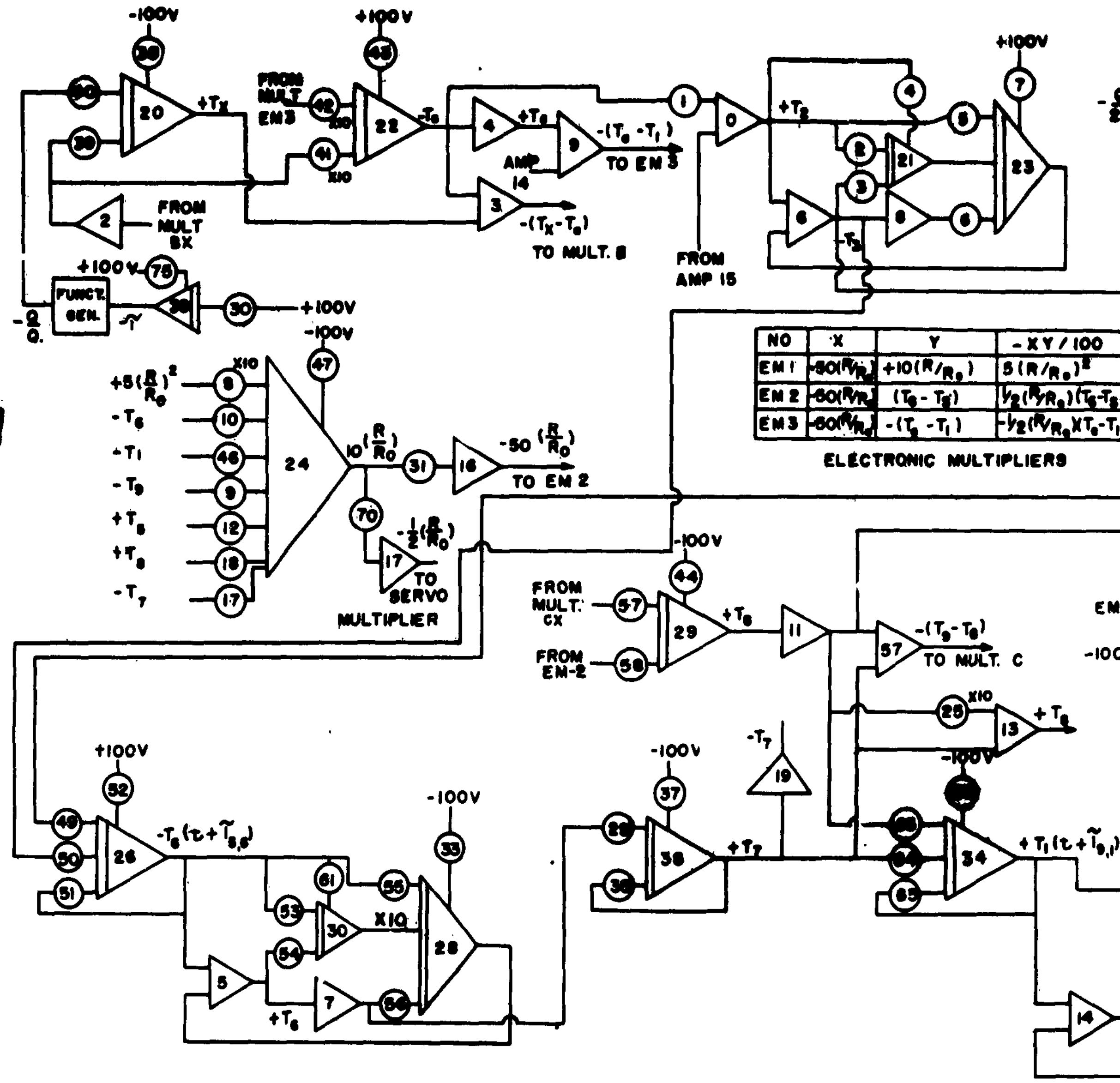


$D_1 = 0.750$
$D_2 = 2.581$
$D_3 = 3.125$
$L_R = 1.831$
$L_G = 11.15$

Fig. 5.3-2 Vertical Schematic Of The Sm-2

Figure 5-9

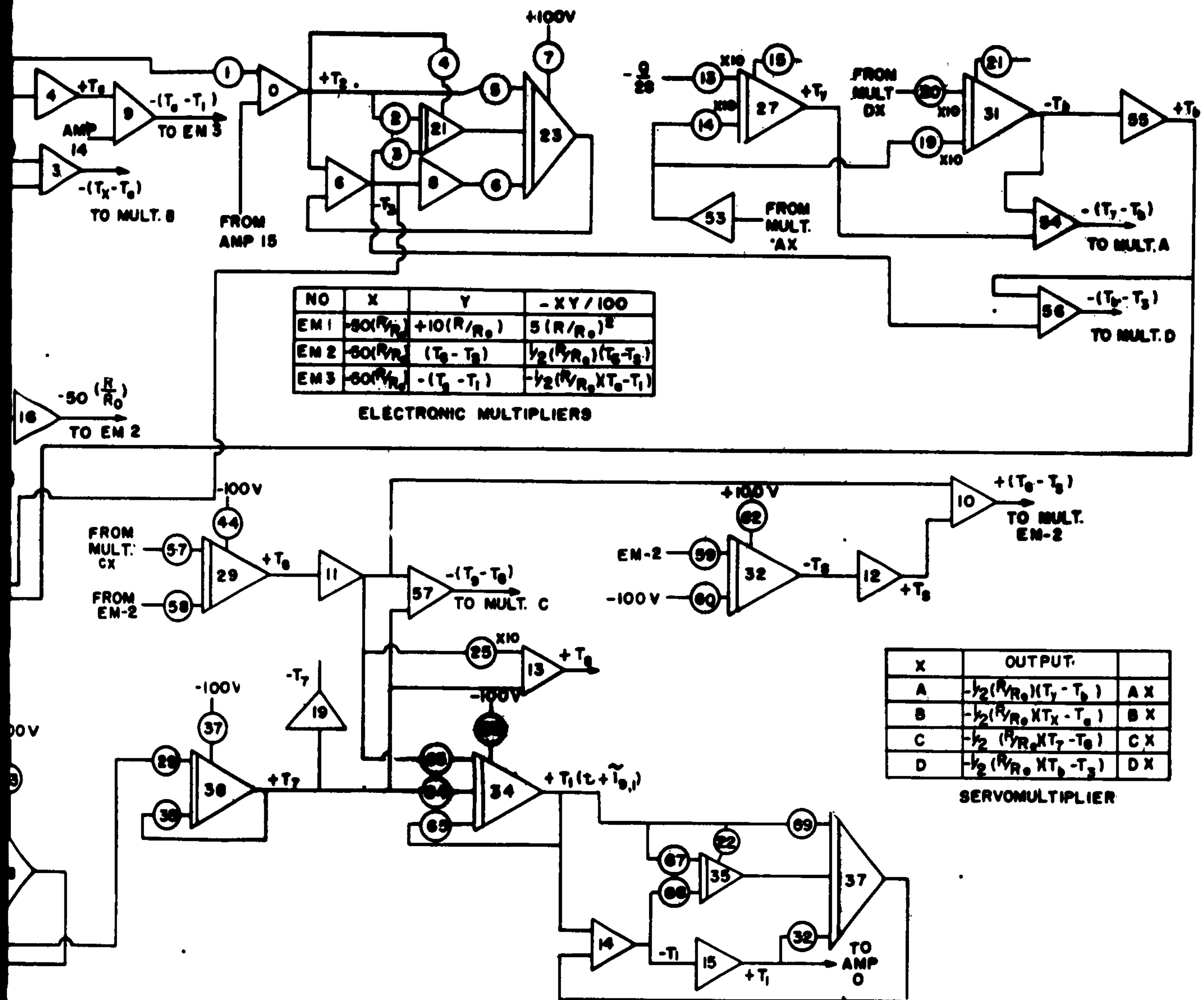
1



NO	X	Y	- XY / 100
EM 1	$-50(R/R_0)$	$+10(R/R_0)$	$5(R/R_0)^2$
EM 2	$-50(R/R_0)$	$(T_6 - T_2)$	$1/2(R/R_0)(T_6-T_2)$
EM 3	$-50(R/R_0)$	$-(T_0 - T_1)$	$-1/2(R/R_0)(T_0-T_1)$

ELECTRONIC MULTIPLIERS

FIGURE 5-10 ANALOG CIRCUIT DIAGRAM-SM-2 DECAY HEAT RE



NO	X	Y	- XY / 100
EM 1	$-50(R/R_0)$	$+10(R/R_0)$	$5(R/R_0)^2$
EM 2	$-50(R/R_0)$	$(T_6 - T_8)$	$\frac{1}{2}(R/R_0)(T_6 - T_8)$
EM 3	$-50(R/R_0)$	$-(T_2 - T_1)$	$-\frac{1}{2}(R/R_0)(T_2 - T_1)$

ELECTRONIC MULTIPLIERS

X	OUTPUT	
A	$-\frac{1}{2}(R/R_0)(T_y - T_b)$	A X
B	$-\frac{1}{2}(R/R_0)(T_x - T_e)$	B X
C	$-\frac{1}{2}(R/R_0)(T_7 - T_8)$	C X
D	$-\frac{1}{2}(R/R_0)(T_b - T_3)$	D X

SERVOMULTIPLIER

FIGURE 5-10 ANALOG CIRCUIT DIAGRAM-SM-2 DECAY HEAT REMOVAL

SM-2 DECAY HEAT REMOVAL WITH 3% OF FULL POWER APPLIED TO SECONDARY SYSTEM, FROZEN PUMP CONDITION

(INITIAL MEAN TEMPERATURE OF COOLANT IN EACH PASS 10°F HIGHER THAN STEADY STATE)

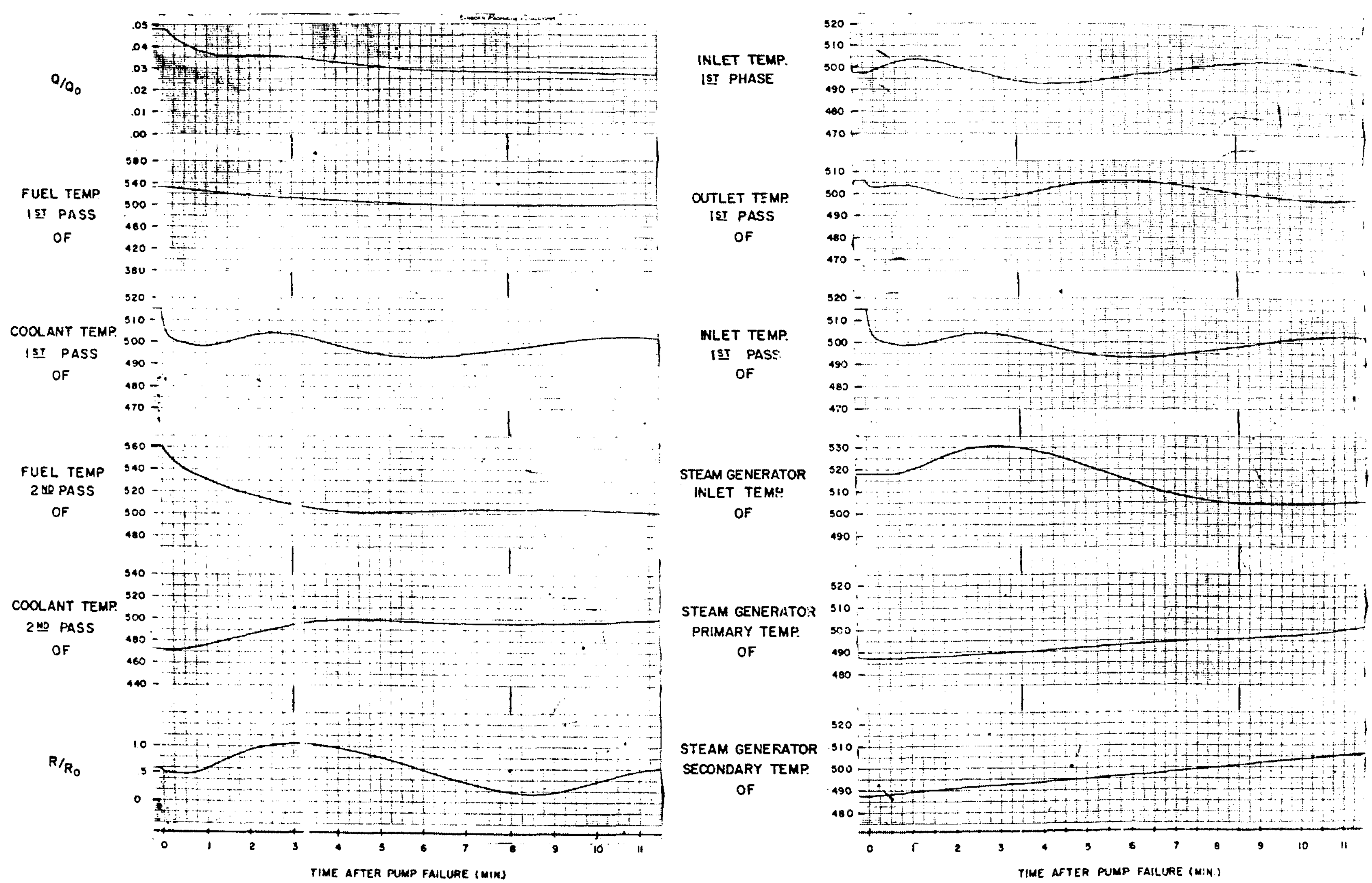


Figure 5-11

SM-2 DECAY HEAT REMOVAL WITH 5% OF FULL POWER APPLIED TO SECONDARY SYSTEM, FROZEN PUMP CONDITION

(INITIAL MEAN TEMPERATURE OF COOLANT IN EACH PASS 10°F HIGHER THAN STEADY STATE)

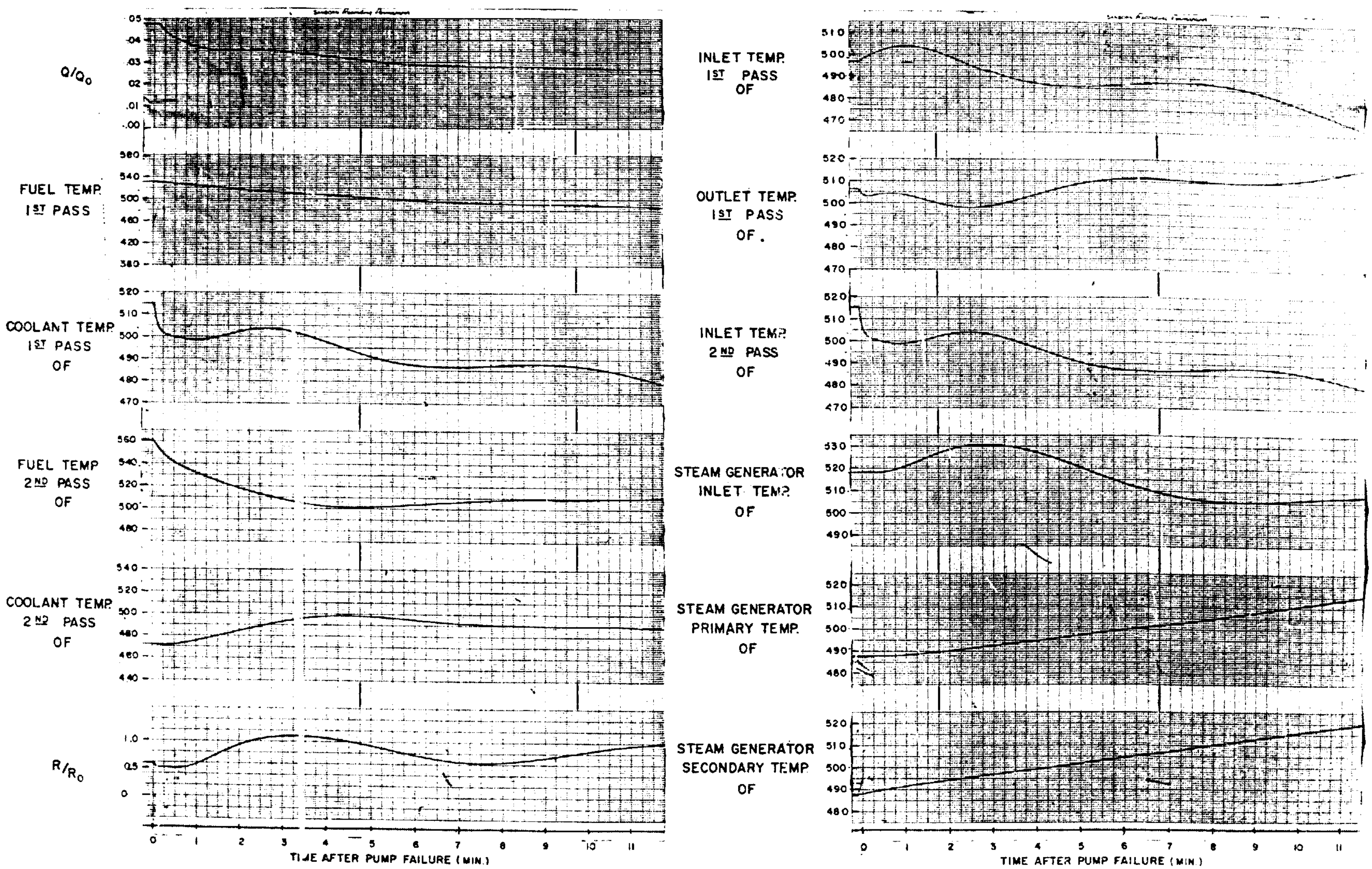


Figure 5-12

6.0 VAPOR CONTAINER

6.1 SELECTION OF CONTAINER CONCEPT

The choice of the vapor container type and size resulted from the study presented in APAE No. 63[1]. This study compared a number of feasible vapor container concepts with respect to cost, space requirements of equipment, maximum internal pressure after the maximum credible accident, secondary shielding, fuel handling, ASME Code requirements, ease of maintenance, ease of construction, ease of access, and ease of inspection and testing. The comparison led to the selection of a 20 ft ID vertical cylinder with ellipsoidal base and hemispherical top.

6.2 VAPOR CONTAINER DESCRIPTION

The construction material of the cylindrical container will be SA-212, Grade B steel. The vapor container will be partly below natural grade level for approximately three-eighths of its height and will be supported by a concrete foundation as shown in Dwg M 11594-56. The loadings from the primary system equipment within the vapor container is transferred evenly to this foundation through a concrete foundation which fills the elliptical bottom of the vapor container.

The overall height of the vapor container will be 55 ft as dictated by the maximum internal pressure, the maximum height of the primary shield tank and the minimum height between shield tank and crane hook as required for the fuel handling tools.

Bracing inside the vapor container will be provided as dictated by final site fuel conditions. Catwalks, supports, and tie-down loads are contemplated as transferred to the shell through this bracing.

Secondary shielding outside the vapor container will be provided by earth backfill which will surround the vapor container from top to foundation (see Dwg M 11594-56). Above natural grade level, the backfill contour will have the appearance of a cone frustum.

The spent fuel tank will be located some distance from the vapor container to allow minimum interference with the earth backfill above natural grade level and to keep tank height to a minimum. The spent fuel tank will be a shop-fabricated unit ready for emplacement and will act as the inside form for a concrete pit. The pit will be covered by a rolling concrete cover. Horizontal transfer of the spent fuel elements from the shield tank inside the vapor container to the spent fuel tank will be accomplished by sled actuation through a 6-in pipe with a manually operated snake as shown on Dwg M 11594-59.

A crib wall or concrete retaining wall will retain the backfill above natural grade in the plane where the backfill would interfere with access to the spent fuel tank and access to the top for a mobile crane.

A 7 ft diameter quick-opening hinged closure is located in the top of the vapor container to allow access to the vapor container from the top for the removal of equipment within the vapor container. Access for maintenance will be through a door at the ground level. A polar crane will be located in the top of the vapor container for internal service.

All service penetrations are sealed and designed to withstand the maximum internal pressure. Electrical penetrations are shown on Dwg M 11594-74.

6.3 VAPOR CONTAINER FOUNDATION AND DRAINAGE

Soil bearing values, ground water level and excavation conditions have been assumed as favorable both for construction and plant economy. The assumed site would drain toward the lake shore about 500 ft distant with a favorable natural slope. Footing and foundation designs contemplate a minimum bearing value of the soil of 4000 lb/ft^2. Foundation drainage and protection from ground water pressure would be secured through the use of gravel filled gravity flow drains. All foundations would be membrane or mopped asphalt coating, water proofed on the outside. Perforated drain tiles would be used in the bottoms of all gravel filled drainage media. All wall and foundation drains would tie into outfalls. Basement flows would be poured over a gravel blanket with drainage provided as above.

Should test borings of the building site indicate rock, an alternate substructure design and drainage plan would be necessary. If the site conditions are such that good drainage of the backfill surrounding the vapor container cannot be obtained, the vapor container shell must be protected. This will be done by erecting a cylindrical concrete retainer wall concentric with the cylindrical vapor container shell, and integral with the concrete foundation of the vapor container. The concrete retainer wall will then retain the backfill rather than the vapor container shell. No major changes would be required to the vapor container itself.

Access openings, pipe penetrations and spent fuel pit can stay in approximately the same location (Dwg. M 11594-56).

6.4 SECONDARY SHIELDING MATERIALS

Secondary shielding for the SM-2 will be provided by concrete and earth backfill. The vapor container will be partly below natural grade level. The reactor will be below the natural grade level. Earth backfill will surround the vapor container from top to bottom (Dwg. M 11594-56).

The secondary shield is designed to permit no more than 0.75 mr/hr exterior to the vapor container during normal full power operation. There will also be sufficient secondary shielding to limit the dose rate to 7.5 mr/hr outside an exclusion area after a maximum credible accident.

Except for the top opening and the connection with the demineralizer room there is a minimum of 8 ft of dirt. This is sufficient shielding to lower the dose rate to 7.5 mr/hr after a maximum credible accident. The top of the vapor container will be shielded by a removable concrete cover. A concrete wall between vapor container and demineralizer room will lower the dose rate to 7.5 mr/hr after a maximum credible accident.

6.5 VAPOR CONTAINER INTEGRITY

The function of the vapor container is to contain the fluids and radioactive materials released from the primary and secondary system components upon a maximum credible accident.

The vapor container shell has to be designed to withstand the internal peak pressure and temperature which will develop inside the vapor container upon release of the pressurized high temperature fluids from the system components after the maximum credible accident.

For the computation of the internal peak pressure and the required shell thickness to withstand this pressure, the following accident conditions have been postulated.

A maximum credible accident has been defined which assumes simultaneous rupture of the primary and secondary system. This rupture is assumed to occur during idling conditions of the plant, when the temperature of the fluids in primary and secondary system are equal. A maximum probable accident has been defined, which assumes rupture of the primary system only during idling conditions of the plant.

The ASME Code allowable stress of 17,500 psi for SA-212, Grade B steel shall not be exceeded for the internal pressure experienced during this maximum probable accident. For the maximum credible accident conditions, the code stress shall not exceed 80 percent of the minimum yield strength of the plate material (35,000 psi for SA-212, Grade B steel).

It is felt that since the loading as caused by the maximum credible accident will occur only once during the lifetime of the container, the allowable stress can be safely fixed at a higher percentage of the minimum yield strength of the plate material.

The volumes available for expansion of the fluids within the container have been calculated for the maximum probable accident and maximum credible accident, and appear in Volume II. After determination of the weight and thermal conditions of the fluids in the system, the specific volume and peak pressure of the fluids after the accident can be determined, assuming constant internal energy before and after the accident. The calculation method fully described in AP Note 78[2] has been used to determine the internal peak pressure.

The air trapped in the container during the accident is assumed to be at 120°F and 14.7 psia prior to rupture. An absolute partial vapor pressure P_v is assumed following completion of blowdown. The corresponding equilibrium temperature, T_e, of the liquid, vapor, and air can be found from the steam tables. The partial air pressure can then be determined from the relationship

$$P_a = P_o \frac{T_e}{T_o} \qquad (1)$$

where P_o and T_o are the original container absolute pressure and temperature. The total peak pressure is found by adding the two partial pressures.

The total water weight (W) and the energy (U), contained in the system (s) are known. Division of the energy by the weight yield is the average water energy per pound (U_m).

The mixture quality can be determined from

$$U_m = U_f + X\ U_{fg} \qquad (2)$$

The specific volume, V_m of the mixture follows from

$$V_m = V_f + XV_{fg} \qquad (3)$$

Since the specific volume of the mixture is known from the start, the trial value can be compared with the actual value. The peak pressure then can be found from a graphical presentation of peak total pressure versus specific volume for various values of average internal energy. Calculations for the SM-2 vapor container are presented in Volume II. A tabulation of the important data follows.

SM-2 VAPOR CONTAINER DATA

Internal diameter, ft	20
Total height, ft	55.5
Required plate thickness for internal peak pressure, in	
Cylindrical section	1.0
Hemispherical section	0.5
Ellipsoidal section	1.0
Gross volume, ft^3	15710
Net volume, ft^3	

SM-2 VAPOR CONTAINER DATA (Cont'd)

After maximum probable accident	11803
After maximum credible accident	11940
Internal energy, Btu $(10)^6$	
After maximum probable accident	4.520
After maximum credible accident	6.882
Peak pressure, psia	
After maximum probable accident	112
After maximum probable accident	140
Temperature, ^{o}F	
After maximum probable accident	332
After maximum credible accident	342
Plate material	SA-212, Grade B
Allowable code stress, psi	17,500
Minimum yield strength, psi	35,000

The vapor container also has to be designed for external loadings. Since the actual site conditions are not known at this time, the external loading cannot be established. The vapor container as designed for internal peak pressure will be braced inside the shell against external loadings as dictated by the site conditions.

6.6 MISSILE PROTECTION PHILOSOPHY

The missile protection philosophy is based on the selection of objects which have potential of becoming missiles, analysis of the level of kinetic energy which may be imparted to the objects, and protection against penetration of the vapor container by those objects which can attain sufficient kinetic energy to become penetration hazards. The calculations of missile velocities are contained in Volume II of this report.

6.6.1 Selection of Possible Missiles

Any object attached to primary piping and vessels and exposed to high pressure coolant fluid may become a missile. The primary vessels may also develop missile potential. Objects selected for analysis as missiles include the following:

a) Primary vessels - reactor, pressurizer, and steam generator

b) Reactor and pressurizer covers

c) Safety valves on the pressurizer

d) Electric heaters in the pressurizer

e) Thermocouple and resistance bulb wells

f) Plugged weldolets and threadolets

g) Blanked drain connections

h) Level controls and pressure connections

6.6.2 Calculation of Kinetic Energy of Missiles

The energy imparted to a missile by an impulse thrust is expressed mathematically by:

$$KE = \int_0^S M_r (U_j - V_m) \, ds \qquad (1)$$

where

KE	=	kinetic energy of the missile, ft-lb
M_r	=	mass flow rate of fluid impinging on the missile, slugs/sec
U_j	=	local jet velocity, ft/sec
V_m	=	local missile velocity, ft/sec
S	=	distance traveled, ft

The mass flow rate and local jet velocity depend upon the size of opening for the jet and the thermodynamic process of the escaping fluid. The calculations are based on an isentropic expansion of fluid at a total jet angle of 60°. It is assumed that there is no slip between steam and water droplets in the jet. The missile is assumed to remain in contact with the jet at least as far as the distance required for the jet to expand to atmospheric pressure. Maximum velocity of the missile occurs when all the energy of the missile is kinetic energy of translation with no rotation. These conditions are met when the missile exposes minimum projected area to the jet stream. When an area larger than the minimum is exposed to the impulse of the yet, instability will cause rotation of the missile. In order to simplify the calculations and to give a conservative evaluation, the area at the throat of the jet was taken equal to the minimum projected area of the missile. In all cases, the missile velocity in (1) may be neglected.

The initial state of the fluid is saturated liquid in the case of expansion from the primary coolant and dry and saturated steam in the case of expansion from the steam volume of the pressurizer. The pressure at the throat of the jet (assumed to be the opening in the vessel or piping) is taken as critical pressure equal to 0.58 of the initial pressure. The condition of state of the fluid is calculated from steam properties. From these properties, the jet velocity is determined by the enthalpy drop for an isentropic expansion by:

$$u_j = 223.7 \sqrt{h_o - h_i} \qquad (2)$$

where h_o is the initial enthalpy and h_i is the isentropic enthalpy at any given point in the jet. The mass flow rate from the throat is:

$$M_r = \rho U_t A_t \qquad (3)$$

where ρ is the throat density in slugs/ft^3, and u_t and A_t are velocity at throat and area of the throat. The mass of fluid impinging on the missile is:

$$M_r = m_r \frac{A_p}{A_j} \qquad (4)$$

where m_r is the mass flow rate at the throat, A_p is the minimum projected area of the missile (assumed equal to the throat area of the jet), and A_j is the local jet cross-sectional area which increases in a geometric relationship to throat area. The ratio of A_p/A_j and the local jet velocity, u_j, thus become unique functions of distance from the throat of the jet. Unfortunately, the total function indicated in (1) is not readily integrable. The use of Simpson's Rule can give a very close approximation, and this method is used in the analysis.

In the case of reaction thrust, the potential kinetic energy is taken as the total available energy of the fluid within the missile for the fluid expanding isentropically from initial state to atmospheric pressure.

When the jet opening in a primary vessel is large compared to the vessel itself, a sudden decompression will result. The methods of shock hydrodynamics are used to calculate jet velocity. The fluid expands isentropically across a rarefaction shock wave. The fluid velocity is given by:

$$U = - \int_{P_o}^{P_r} \frac{1}{\rho} \sqrt{\frac{d\rho}{dP}} \, dP \qquad (5)$$

where u is jet velocity, ρ is fluid density, P_o is initial fluid pressure and P_r is the pressure after the rarefaction wave. A compression wave is propagated outward in atmospheric air and the particle velocity of air is given by:

$$U_s = C_o \sqrt{\frac{2}{\gamma}} \frac{\left(\frac{P_r}{P_a} - 1\right)}{\sqrt{(\gamma + 1)\left(\frac{P_r}{P_a}\right) + (\gamma - 1)}} \qquad (6)$$

where u_s is air velocity, C_o is local speed of sound in air, γ for air is 1.4, P_r is the pressure of the compression wave equal to the pressure of the rarefaction wave, and P_a is atmospheric pressure. At the interface between the jet fluid and air the velocities must be equal; that is, (5) and (6) define identical velocities. If the density of the fluid is expressed as a function of pressure ratio of the shock wave and for the pressure P_r, the resulting solution for P_r may then be used in (5) to solve for jet velocity. The methods described above are then used to calculate missile kinetic energy.

6.6.3 Protection Against Vapor Container Penetration

The ability of a missile to penetrate the vapor container depends upon its kinetic energy at impact, its shape, and the area involved at the point of impact. Where these factors have combined to present a penetration hazard, protection against penetration must be taken either by shielding the exposed area of impact or by holddown of the objects to prevent missile action.

Penetration of the steel shell which forms the vapor container is based on the Stanford Research Institute formula for penetration using rod-like missiles. The equation defining energy for penetration of mild steel is:

$$e = 16500\ T^2 + 1500\ T \qquad (7)$$

where e is the missile kinetic energy per unit diameter of missile in ft-lbs per inch and T is wall thickness of the container in inches. A conservative approach is used by assuming that the missile strikes the container at its minimum diameter so that the value of e in (7) is a maximum for a given missile.

6.6.4 Protection Against Small Objects as Missiles

In no case did the analysis indicate that small objects present a penetration hazard. The following tabulation summarizes calculated missile velocities and the corresponding velocity for penetration of the vapor container.

COMPARISON OF MISSILE VELOCITY TO PENETRATION VELOCITY

Item	Weight (lb)	Actual Velocity ft/sec	Penetration Velocity ft/sec
Thermocouple and well	5	70	580
Thermocouple well	2.5	100	820
Plugged threadolet	1	160	1300
Heater in pressurizer	5.7	60	540
Safety valve on pressurizer	170	80	100
Pressurizer cover	600	140	*220

* May be 140 ft/sec depending on way in which the cover strikes the vapor container.

Several other small objects such as the drain valves, level controls, and pressure connections are comparable in diameter and weight to those shown in the table above.

The reactor cover is not considered as a missile hazard. Simultaneous failure of all bolts is not a realistic assumption; sequential bolt failure would release the pressure and tend to deflect the cover out of the jet path, and the shield water above the cover would exert a very large viscous drag on the cover. Other considerations, such as partial collapse of the jet as it strikes the cold water of the shield and the inertia of the mass of shield water, also minimize missile hazard potential.

6.6.5 Protection Against Primary Vessels as Missiles

The pressurizer vessel, reactor, and steam generator will be subjected to large thrust forces if the vessels failed at their largest diameters. Consequently these vessels will be secured against missile action.

The maximum static thrust of the reactor due to failure at its largest diameter was calculated as $8.5\ (10)^5$ lb. The vessel will be secured against this force by the radiation shields which are welded to the base plate.

The maximum thrust of the steam generator was found to be $9.3\ (10)^5$ lb. This vessel will be secured against this force by hold-down devices to the base plate and by spring-loaded devices attached to the vapor container.

The pressurizer can develop a total force of $2.244\ (10)^6$ lb. The retaining members will be designed for this force.

REFERENCES

1. Celentano, M. J., and Van Kessel, H. F., "Vapor Container Concept Study for SM-2".

2 Brondel, J. O., "Volume Requirements for Single Chamber Vapor Containers", AP Note 78, September 25, 1957.

7.0 SKIDS

7.1 DESCRIPTION

The primary system skid and the primary system auxiliary skid are designed as shop assembled units representing the maximum combination of components to form a single package meeting shipping size limitations of high pressure complex equipment. This configuration of structure is dictated by the specific requirements of the components attached, the shipping loading conditions of 8G's fore and aft, 8G's vertical, and 5G's lateral, the installed loading and operating conditions and the seismic dynamic loading forces of the plant site. The primary skid weighs 99,464 lbs; the auxiliary skid 27,500 lbs.

7.2 PRIMARY SKID (Dwgs. M-11594-65 and M-11594-66)

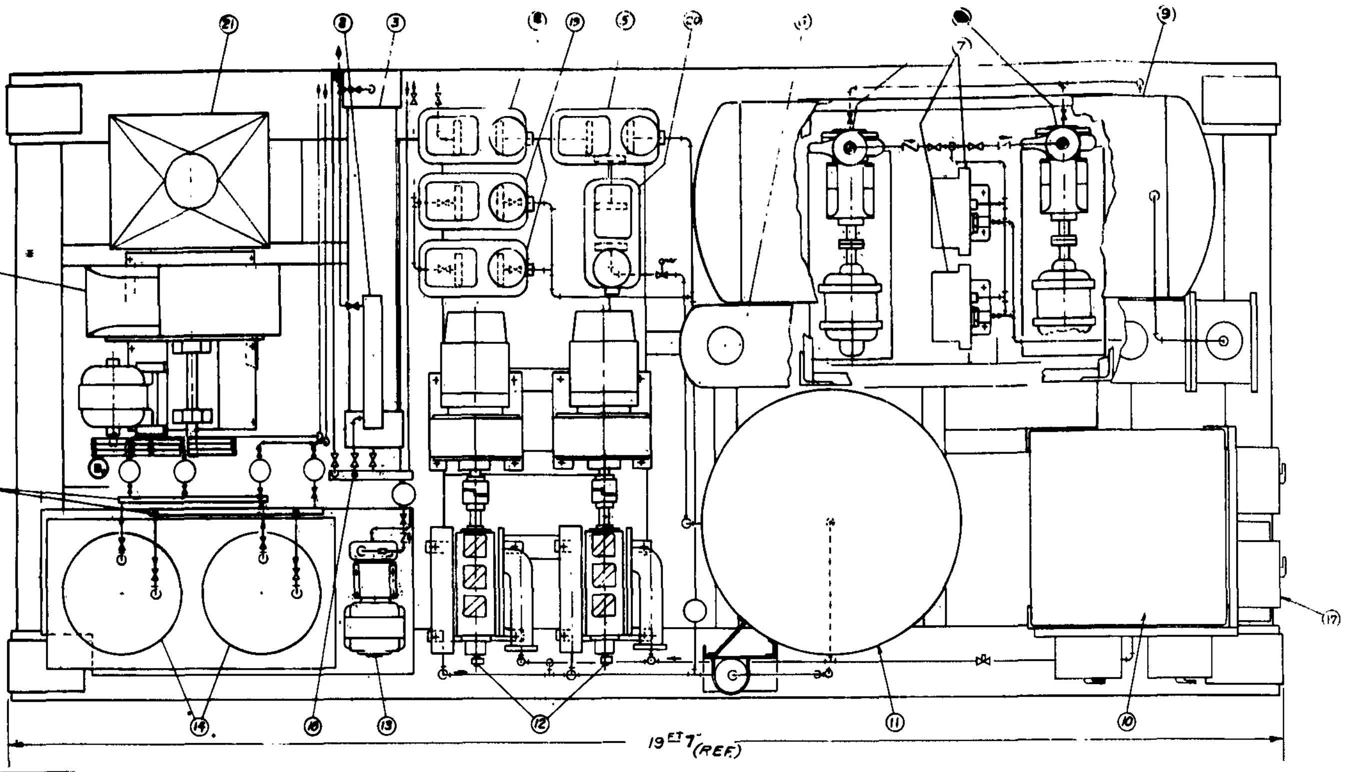
The primary skid is placed inside the vapor container and forms the basic unit of the power complex in that the reactor, lower and upper shield tanks, control rod drives, steam generator shielding, primary piping, and primary coolant pump are all mounted on this structure. The skid is secured to the concrete footing with anchor bolts to insure bearing for transfer of the loads to the foundation.

7.3 PRIMARY SYSTEM AUXILIARY SKID

The primary system auxiliaries skid (Dwgs. M-11594-67 and M-11594-64) is located in the demineralizer room. This placement provides the close arrangement to the primary skid within the vapor container and served by the elements on the auxiliary skid.

The auxiliary skid structure is designed as a rigid base for the pumps, tank filters, piping manifolds, heat exchangers, demineralizers, boron tank, control valves and electrical station. This compact and intricate combination of pipes, valves, pumps and tanks provide for ease of operation and service.

[redacted]



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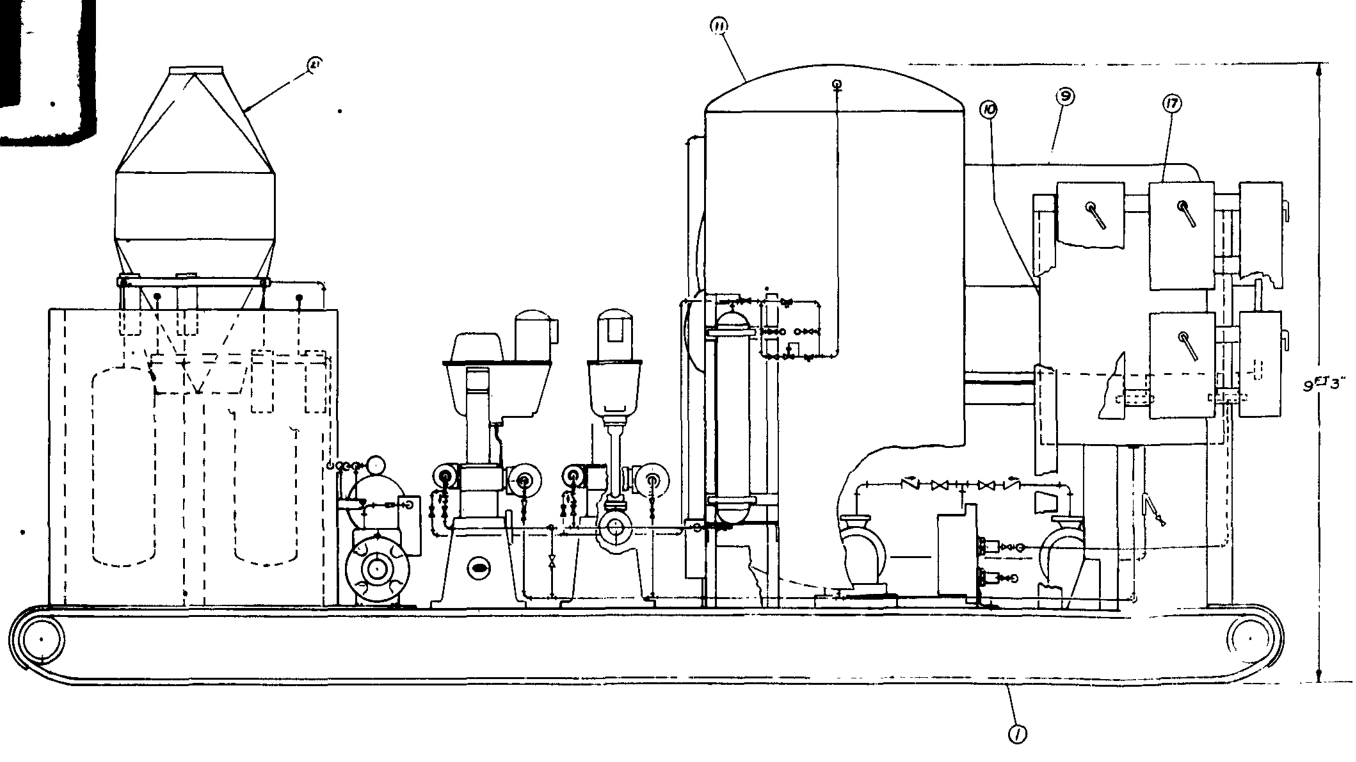
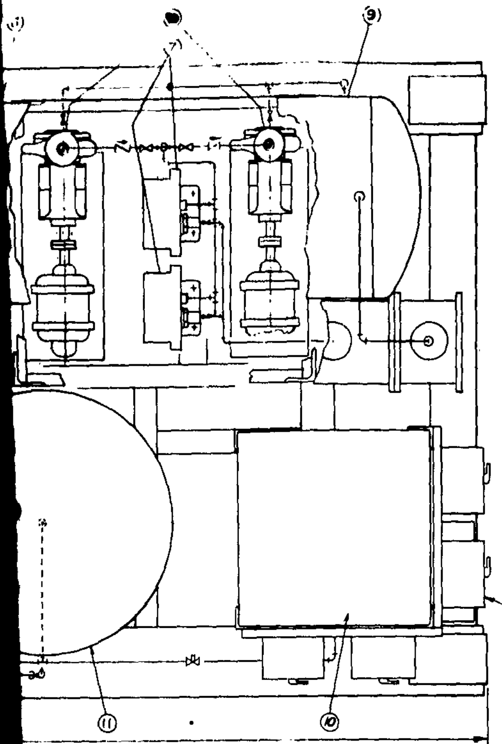
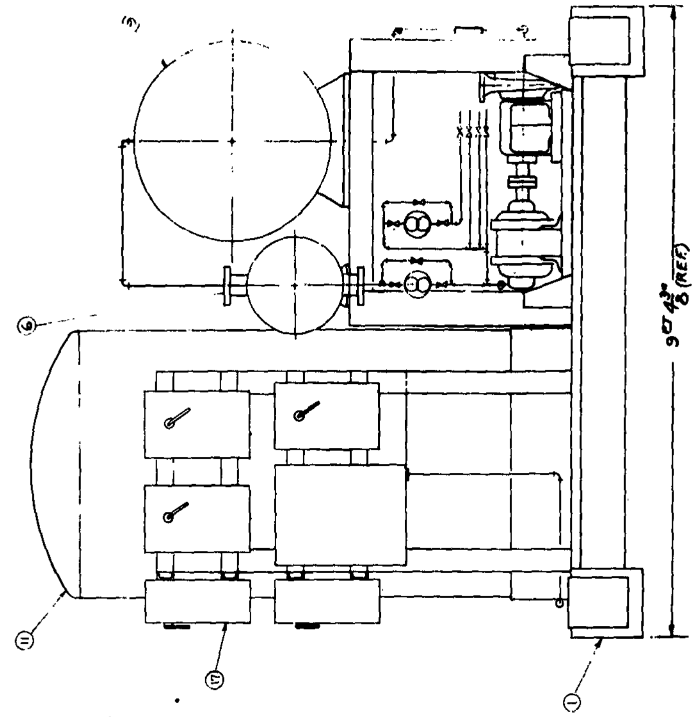
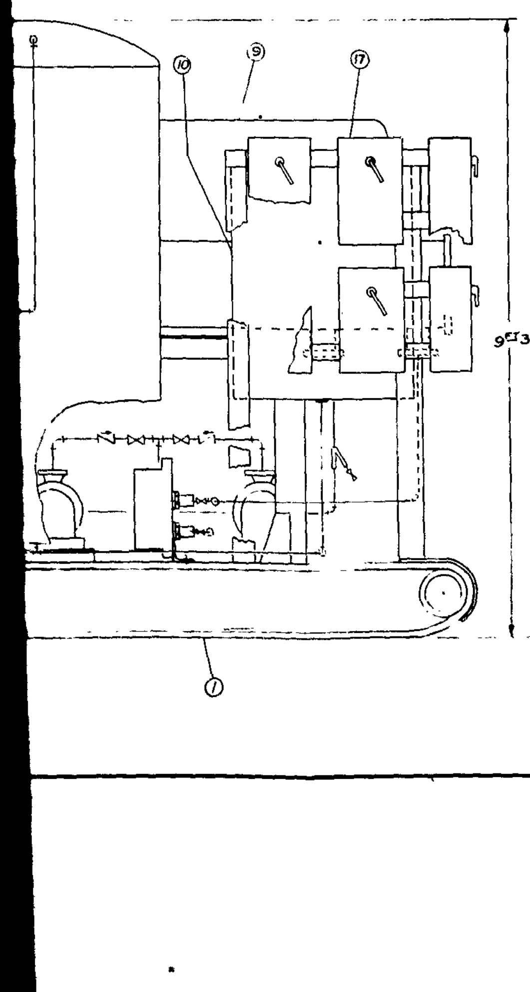


A	COMPLETE REVISION	
B	(1) ADDED DRG. NO. M11594-64 [illegible] (2) ADDED PULLEY TO MOTOR & PULLEY TO BLOWER (PT. 16)	[illegible]





2



OPERATING UNITS —— 12500
PIPING & VALVES —— 5000
SKID & SUPPORTS —— 10000
APPROX TOTAL WT. 27,500 LBS

* DEMINERALIZERS NOT SHIPPED ON SKID

PT. NO.	QTY.	DESCRIPTION		WEIGHT
21	1	V.C. VENTILATING FILTER		800 LBS
20	1	PRIM. MAKE-UP RETURN VALVE		50 LBS
19	2	SEAL WATER REG. VALVES		100 LBS
18	1	SPENT FUEL SYSTEM HEADER		30 LBS
17	1	ELECTRICAL PANEL		900 LBS
16	1	VENTILATING BLOWER		520 LBS
15	2	DEMINERALIZER HEADERS		50 LBS
* 14	2	DEMINERALIZERS		
13	1	SPENT FUEL PIT RECIRC. PUMP		220 LBS
12	2	PRIMARY MAKE-UP PUMPS		3700 LBS
11	1	PRIMARY MAKE-UP TANK		2400 LBS
10	1	BORON TANK		350 LBS
9	1	V.C. COOLING H_2O STORAGE TANK		1300 LBS
8	2	V.C. COOLING WATER PUMPS		700 LBS
7	2	PRIMARY-SEC. B.D. COOLERS		340 LBS
6	1	V.C. COOLING WATER COOLER		700 LBS
5	1	PRIMARY MAKE-UP PROPORT. VALVE		50 LBS
4	1	PRIM. MAKE-UP FLOW CONTROL VALVE		50 LBS
3	1	SPENT FUEL COOLER		170 LBS
2	1	HEATER		60 LBS
1	1	SKID	M11594-67	10000 LBS

LIST OF MATERIAL

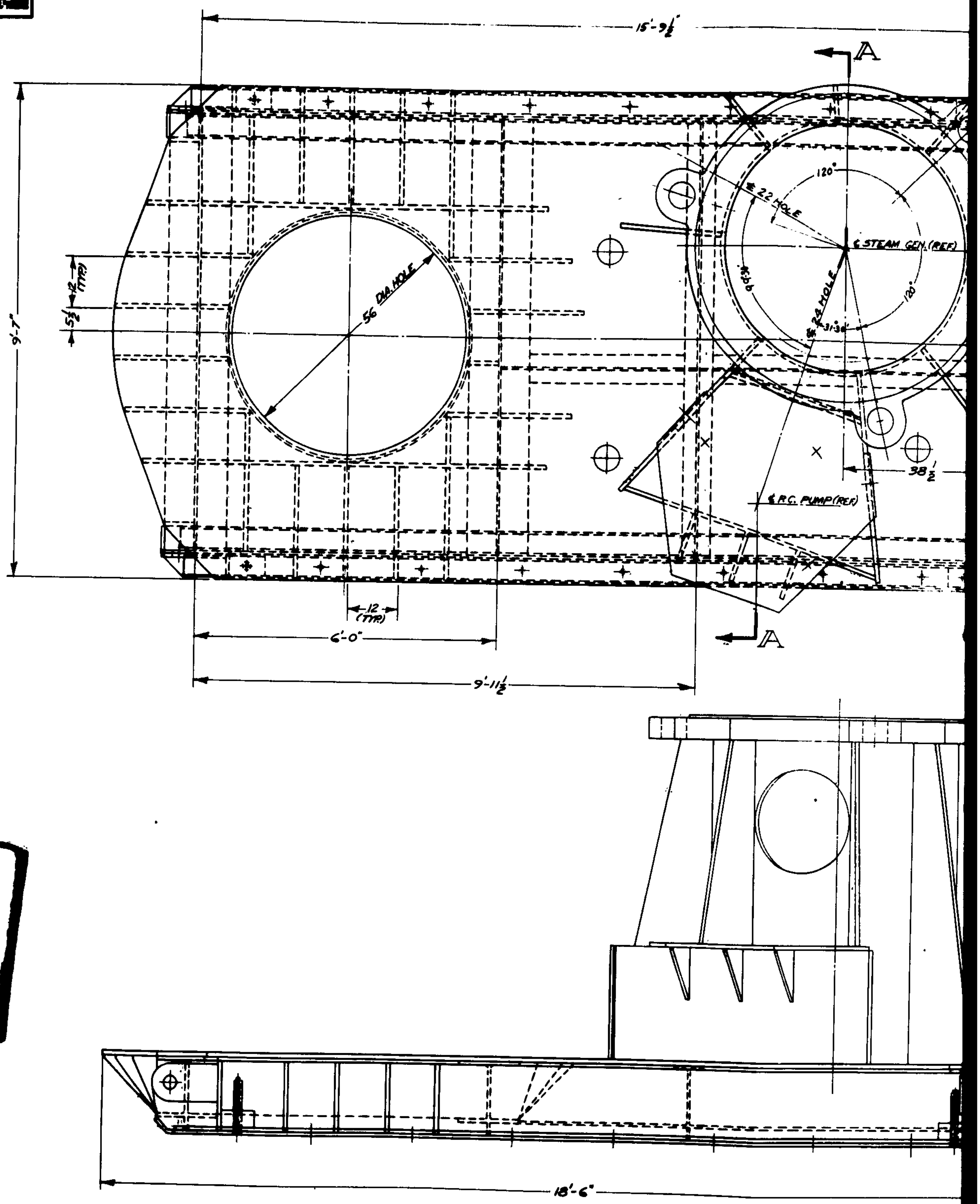
M11594-64 B

PRIMARY SYSTEM, AUX SKID EQUIPMENT ARRANGEMENT

M11594-64

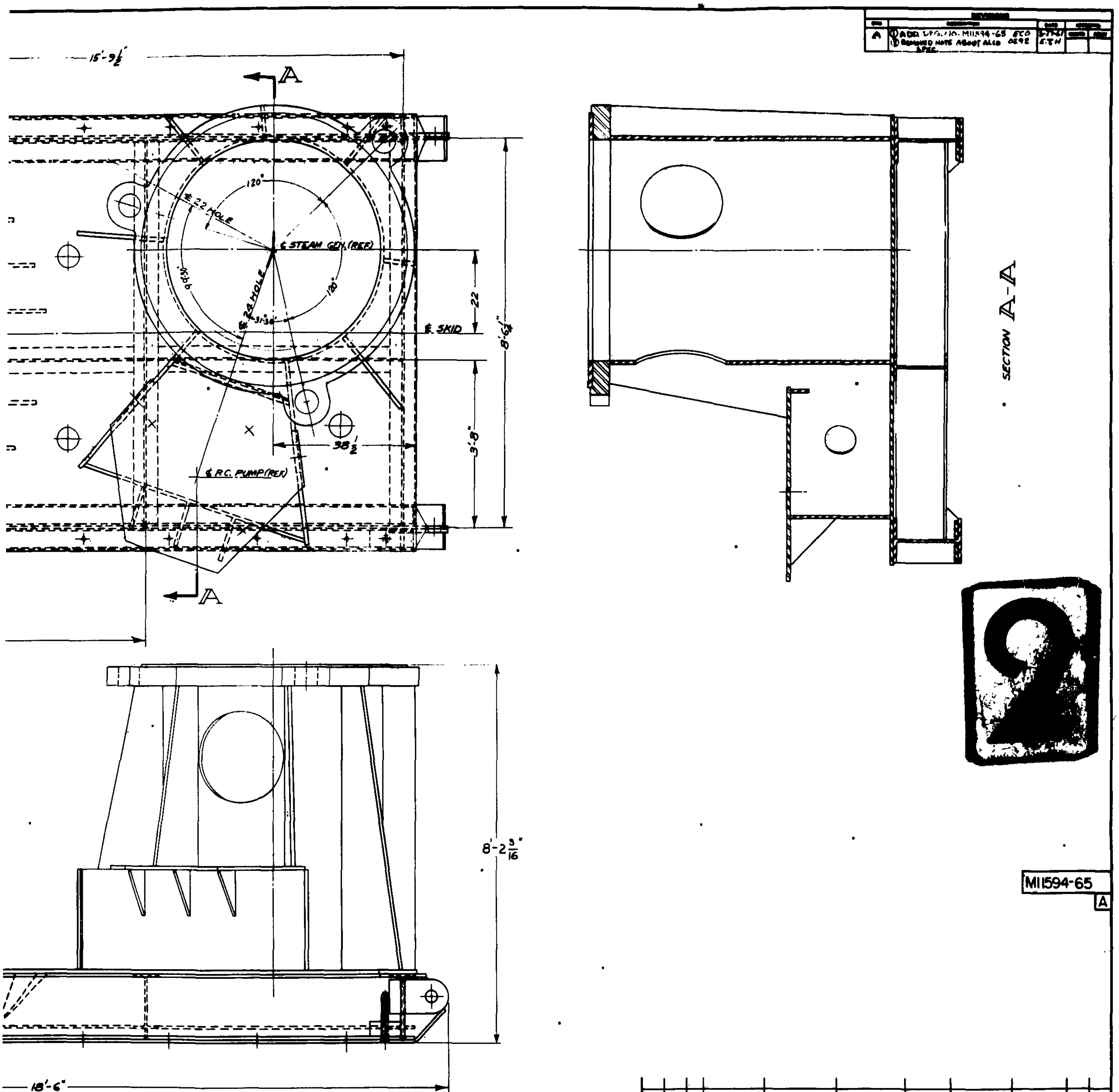
1" : 1'-0"

APPLICATION



1

FABRICATE FROM HIGH STRENGTH
LOW-ALLOY STEEL A.S.T.M. SPEC. A242
MATERIALS AND WELDS SHALL HAVE 15 FT. LBS
KEY HOLE CHARPY MIN.



FABRICATE FROM HIGH STRENGTH
LOW-ALLOY STEEL A.S.T.M. SPEC. A242
MATERIALS AND WELDS SHALL HAVE 15 FT. LBS
KEY HOLE CHARPY MIN.

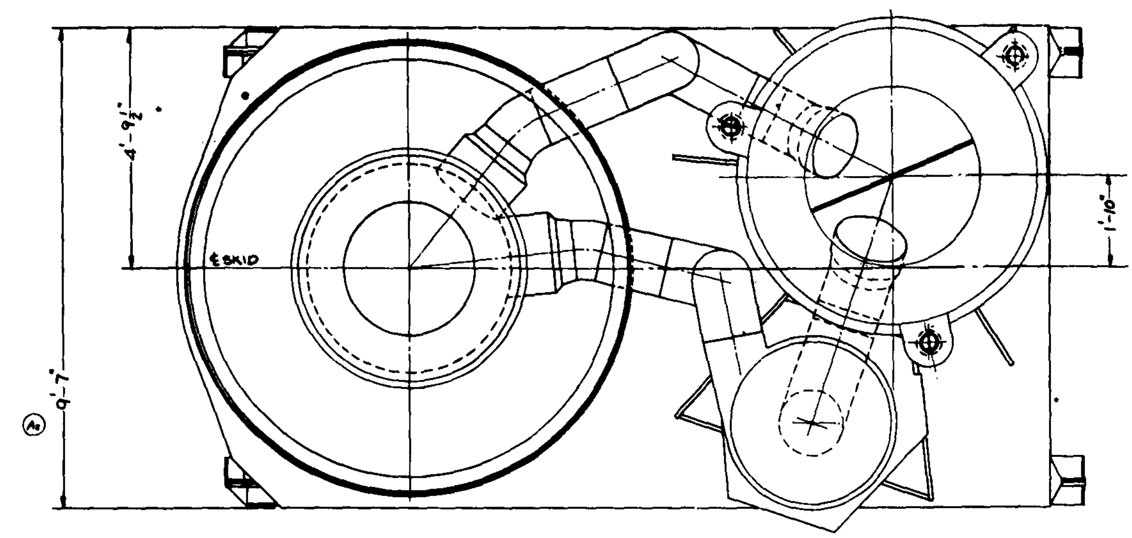
LIST OF MATERIAL

APPLICATION

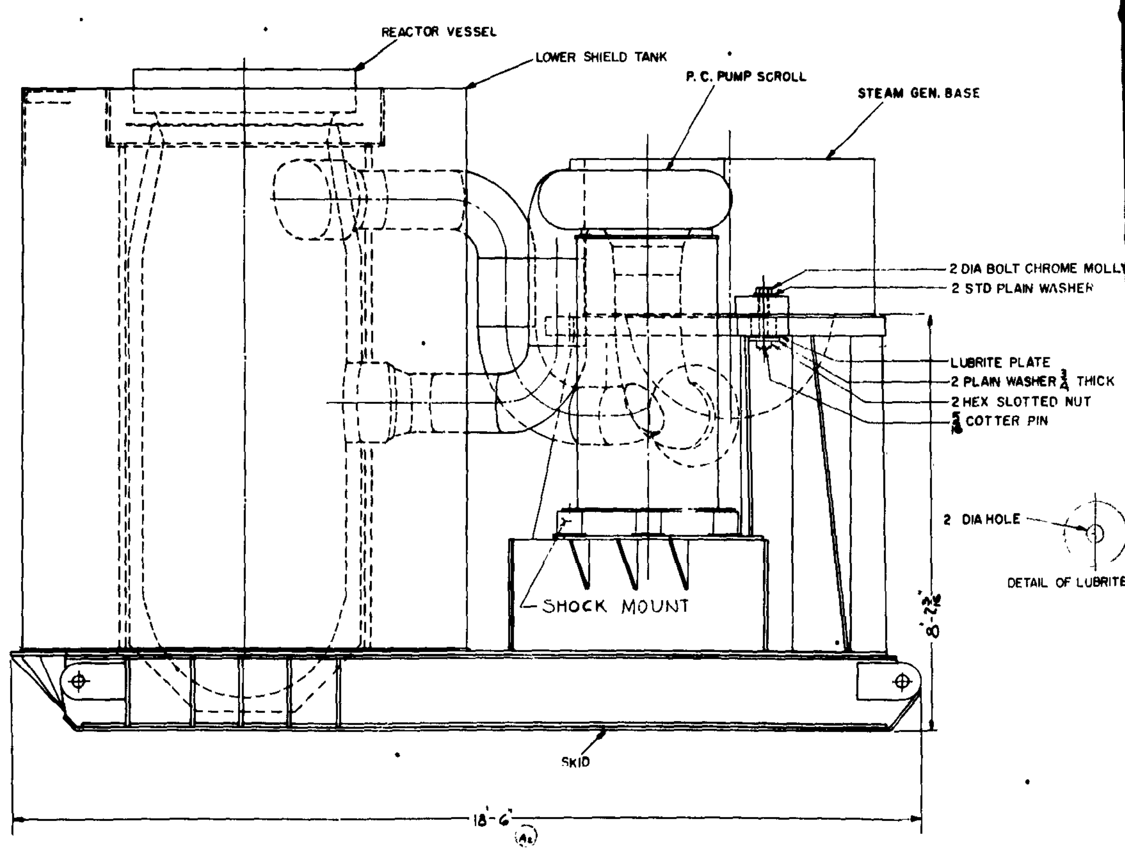
PRIMARY SYSTEM
SKID STRUCTURE

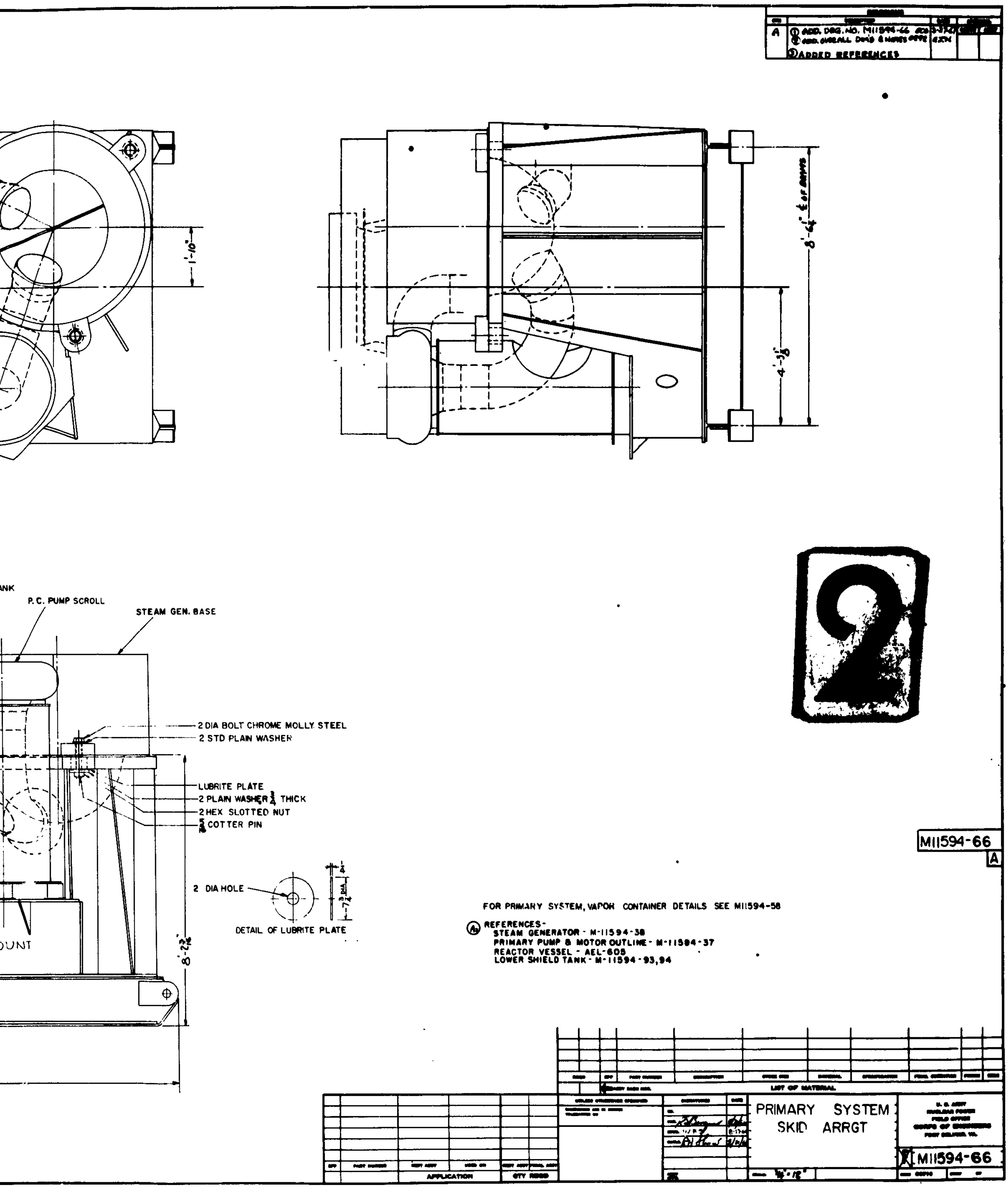
M11594-65

[redacted]


4'-9½"
1'-10"
℄ SKID
9'-7"
A₂

1

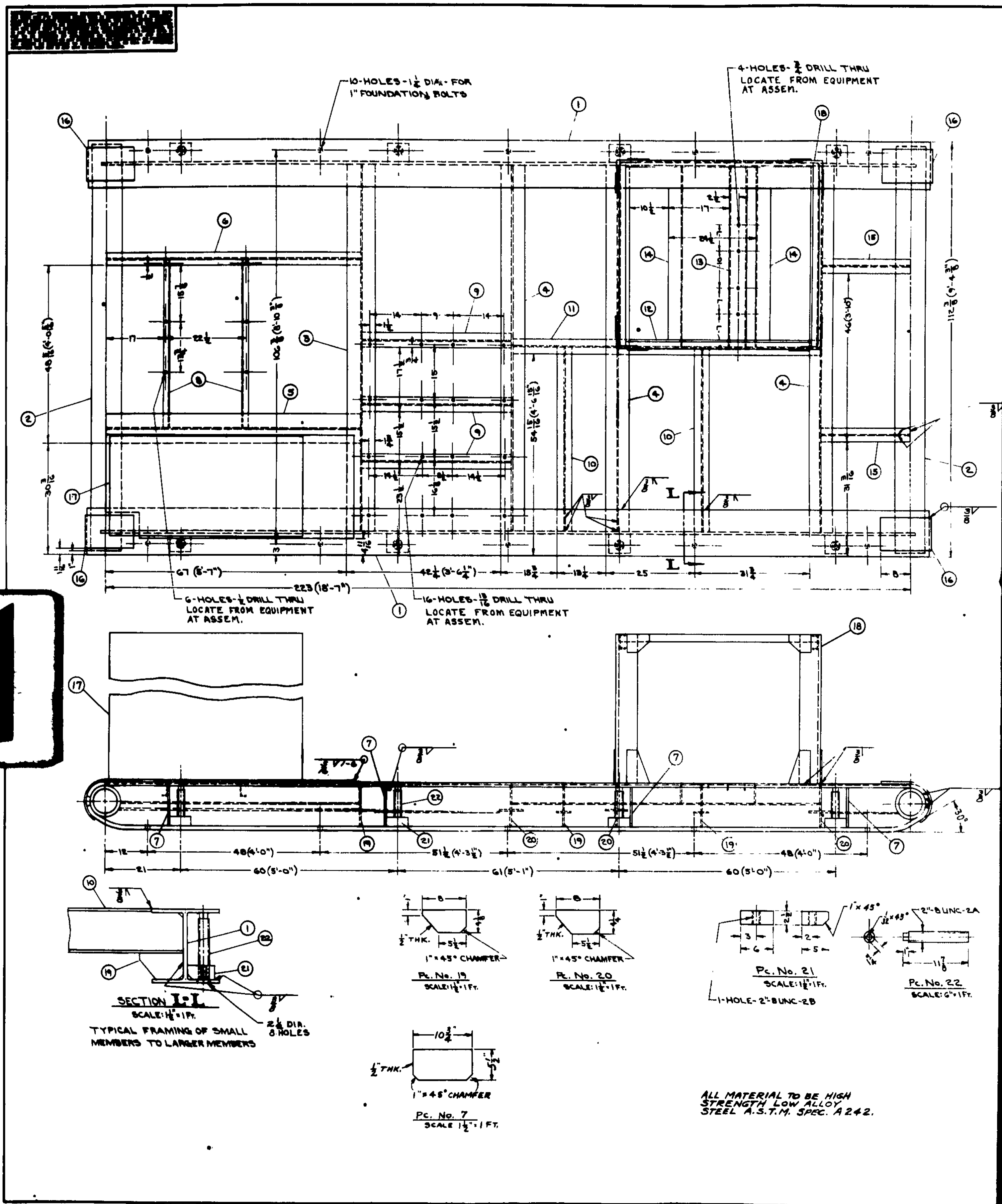

REACTOR VESSEL
LOWER SHIELD TANK
P. C. PUMP SCROLL
STEAM GEN. BASE
2 DIA BOLT CHROME MOLLY
2 STD PLAIN WASHER
LUBRITE PLATE
2 PLAIN WASHER ¾ THICK
2 HEX SLOTTED NUT
⅜ COTTER PIN
2 DIA HOLE
DETAIL OF LUBRITE
SHOCK MOUNT
8'-7 7/16"
SKID
18'-6"
A₂



FOR PRIMARY SYSTEM, VAPOR CONTAINER DETAILS SEE M11594-58

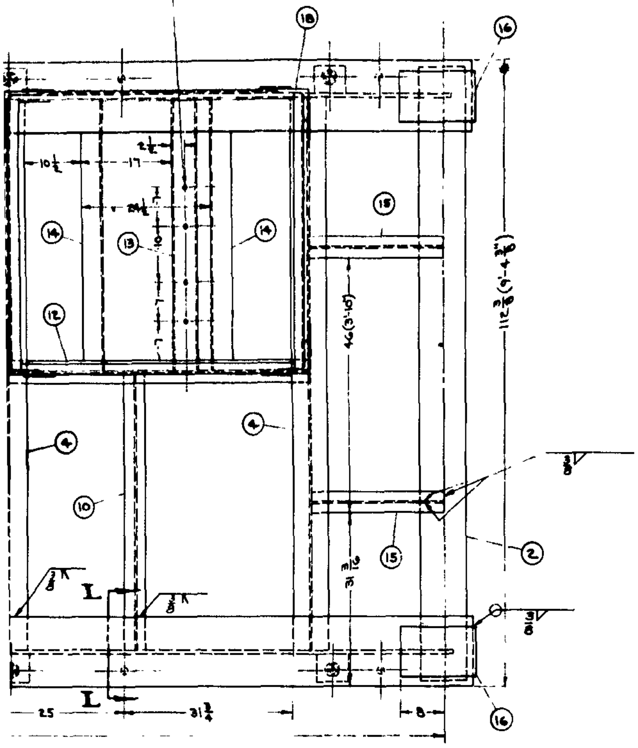
(A3) REFERENCES-
STEAM GENERATOR - M-11594-38
PRIMARY PUMP & MOTOR OUTLINE - M-11594-37
REACTOR VESSEL - AEL-605
LOWER SHIELD TANK - M-11594-93,94

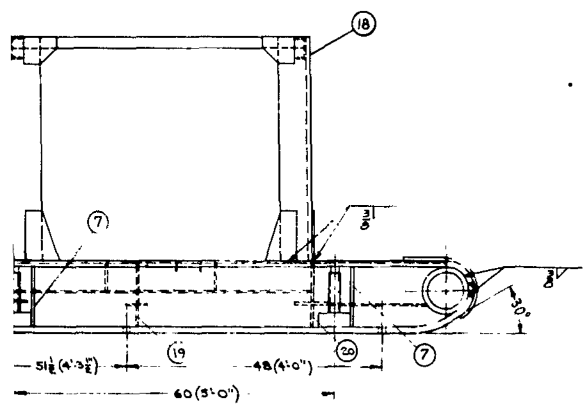
1

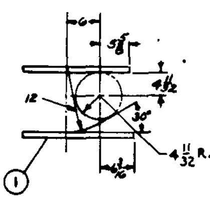

[redacted]
10-HOLES-1 1/8 DIA.- FOR 1" FOUNDATION BOLTS
4-HOLES-7/8 DRILL THRU LOCATE FROM EQUIPMENT AT ASSEM.
67 (5'-7")
223 (18'-7")
6-HOLES-1/2 DRILL THRU LOCATE FROM EQUIPMENT AT ASSEM.
16-HOLES-13/16 DRILL THRU LOCATE FROM EQUIPMENT AT ASSEM.
48 (4'-0")
60 (5'-0")
61 (5'-1")
60 (5'-0")
SECTION L-L
SCALE: 1 1/2"=1 FT.
TYPICAL FRAMING OF SMALL MEMBERS TO LARGER MEMBERS
2 1/4 DIA. 8 HOLES
1"×45° CHAMFER
Pc. No. 19
SCALE: 1 1/2"=1 FT.
Pc. No. 20
SCALE: 1 1/2"=1 FT.
Pc. No. 21
SCALE: 1 1/2"=1 FT.
1-HOLE-2"-8UNC-2B
2"-8UNC-2A
Pc. No. 22
SCALE: 6"=1 FT.
Pc. No. 7
SCALE 1 1/2"=1 FT.
ALL MATERIAL TO BE HIGH STRENGTH LOW ALLOY STEEL A.S.T.M. SPEC. A242.

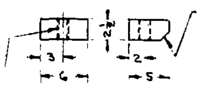
A	① ADD [illegible] ECO	[illegible]		
	② ADD NOTE "REF. ASSY DRG." [illegible]			

4-HOLES-$\frac{3}{4}$ DRILL THRU
LOCATE FROM EQUIPMENT
AT ASSEM.

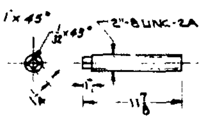








Pc. No. 21
SCALE: $1\frac{1}{2}$"=1 FT.
1-HOLE-2"-8UNC-2B



Pc. No. 22
SCALE: 6"=1 FT.

REF. ASSY DRG. M11594-64

ALL MATERIAL TO BE HIGH STRENGTH LOW ALLOY STEEL A.S.T.M. SPEC. A242.

M11594-67
A

PT. NO.	QUAN.	DESCRIPTION		WEIGHT	MATERIAL	SPECIFICATION
22	8	2" DIA. ROD × $11\frac{1}{2}$" LG		84.5	HIGH STRGTH STL	ASTM A242
21	8	$2\frac{1}{2}$ × 5 PLATE × 6" LG.		169.6	HIGH STRGTH STL	ASTM A242
20	6	[illegible] × 4[illegible] PLATE × 8" LG		28.9	HIGH STRGTH STL	ASTM A242
19	4	[illegible] × 4[illegible] PLATE × 8 LG.		18.7	HIGH STRGTH STL	ASTM A242
18	1	M11594-68	PEDESTAL	518.5		
17	1	M11594-69	BOX	771.2		
16	4	$\frac{1}{2}$ × 9 PLATE × 23[illegible]" LG.		118.5	HIGH STRGTH STL	ASTM A242
15	2	8×4 WF × 13# × 24[illegible]" LG		53.3	HIGH STRGTH STL	ASTM A242
14	2	6×4 L × [illegible] × 48[illegible] (4'-0[illegible]) LG.		83.6	HIGH STRGTH STL	ASTM A242
13	1	5×1[illegible] [× 9# × 48[illegible] (4'-0[illegible]) LG		36.5	HIGH STRGTH STL	ASTM A242
12	1	6×4 WF × 12# × 56[illegible] (4'-8[illegible]") LG		56.5	HIGH STRGTH STL	ASTM A242
11	1	6×4 WF × 12# × 28[illegible]		28.75	HIGH STRGTH STL	ASTM A242
10	2	8×4 WF × 13# × 50[illegible] (4'-2[illegible]") LG		108.8	HIGH STRGTH STL	ASTM A242
9	3	8×4 WF × 13# × 41[illegible] (3'-5[illegible]) LG		133.7	HIGH STRGTH STL	ASTM A242
8	2	4×1[illegible]" [× 5.4# × 45[illegible] (3'-9[illegible]) LG		41.2.	HIGH STRGTH STL	ASTM A242
7	8	[illegible] × 5[illegible] PLATE × 10[illegible] LG.		67.	HIGH STRGTH STL	ASTM A242
6	1	6×3[illegible] I × 12.5# × 70[illegible] (5'-10[illegible]) LG		73.8	HIGH STRGTH STL	ASTM A242
5	1	8×8 WF × 31# × 70[illegible] (5'-10[illegible]) LG		182.9	HIGH STRGTH STL	ASTM A242
4	3	8×6[illegible] WF × 24# × 99[illegible] (8'-3[illegible]) LG		595.5	HIGH STRGTH STL	ASTM A242
3	1	8×8 WF × 31# × 99[illegible] (8'-3[illegible]") LG		256.4	HIGH STRGTH STL	ASTM A-242
2	2	XX PIPE - 8" × 110[illegible] (9'-2[illegible]") LG		1,332.2	STEEL	ASTM A53[illegible]
1	2	12×12 WF × 1[illegible]		5,217.4#	HIGH STRGTH STL	ASTM A-242

LIST OF MATERIAL

APPLICATION | QTY REQD

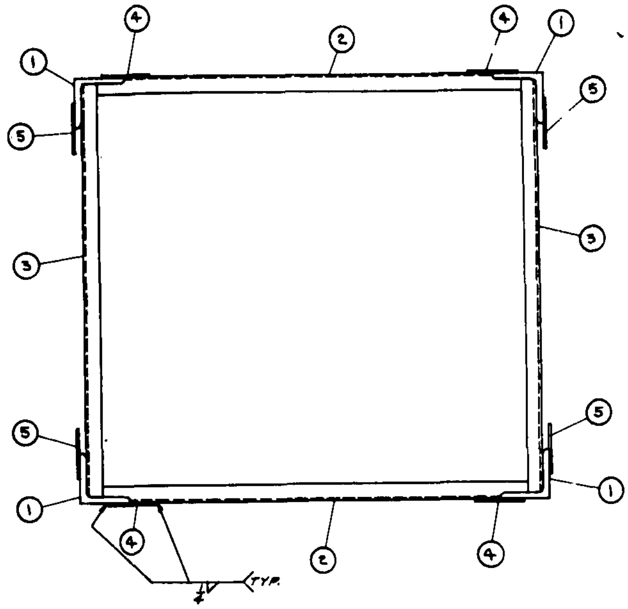
R. LAMPERON
WPS

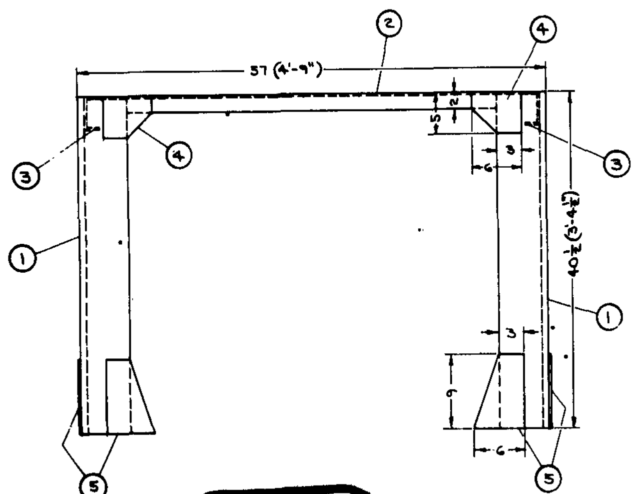
PRIMARY SYSTEM
AUXILLIARY
SKID STRUCTURE

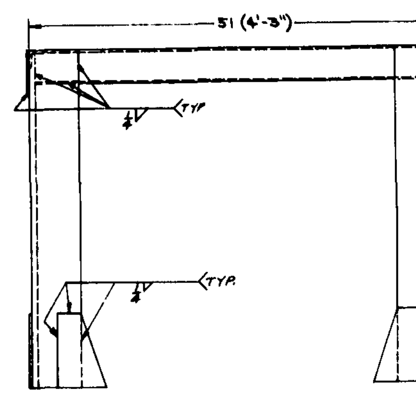
M11594-67

1 1/2 = 1 Ft

[redacted]







1

ALL MATERIAL TO BE HIGH STRENGTH LOW ALLOY STEEL ASTM SPEC.-A-242

SYM	DESCRIPTION		DATE	APPROVAL
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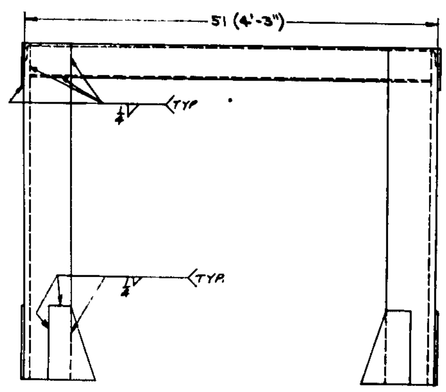


LL MATERIAL TO BE HIGH
TRENGTH LOW ALLOY STEEL
STM SPEC.-A-242

M11594-68
A

REF. DWG. M11594-69
DWG M11594-64

PC. NO.	QUAN.		DESCRIPTION.	WEIGHT	MATERIAL	SPECIFICATION
5	8		[illegible]x6" PLATE - 9" LG.	28.7#	[illegible]	ASTM-A-242
4	4		[illegible]x5" PLATE - 6" LG	9.0#	[illegible]	ASTM-A-242
3	2		4x1[illegible] x7.25# [-49[illegible]" LG (4'-1[illegible])	59.8#	[illegible]	ASTM-A-242
2	2		2x2x[illegible] L - 52" LG. (4'-4")	34.0#	[illegible]	ASTM-A-242
1	4		6x6x[illegible] L - 40[illegible]" LG. (3'-4[illegible])	387.0#	[illegible]	ASTM-A-242

LIST OF MATERIAL

DR. R. LAMPHRON	9-1-60
CH. [illegible]	9-[illegible]
WES	[illegible]
[illegible]	[illegible]
[illegible]	[illegible]

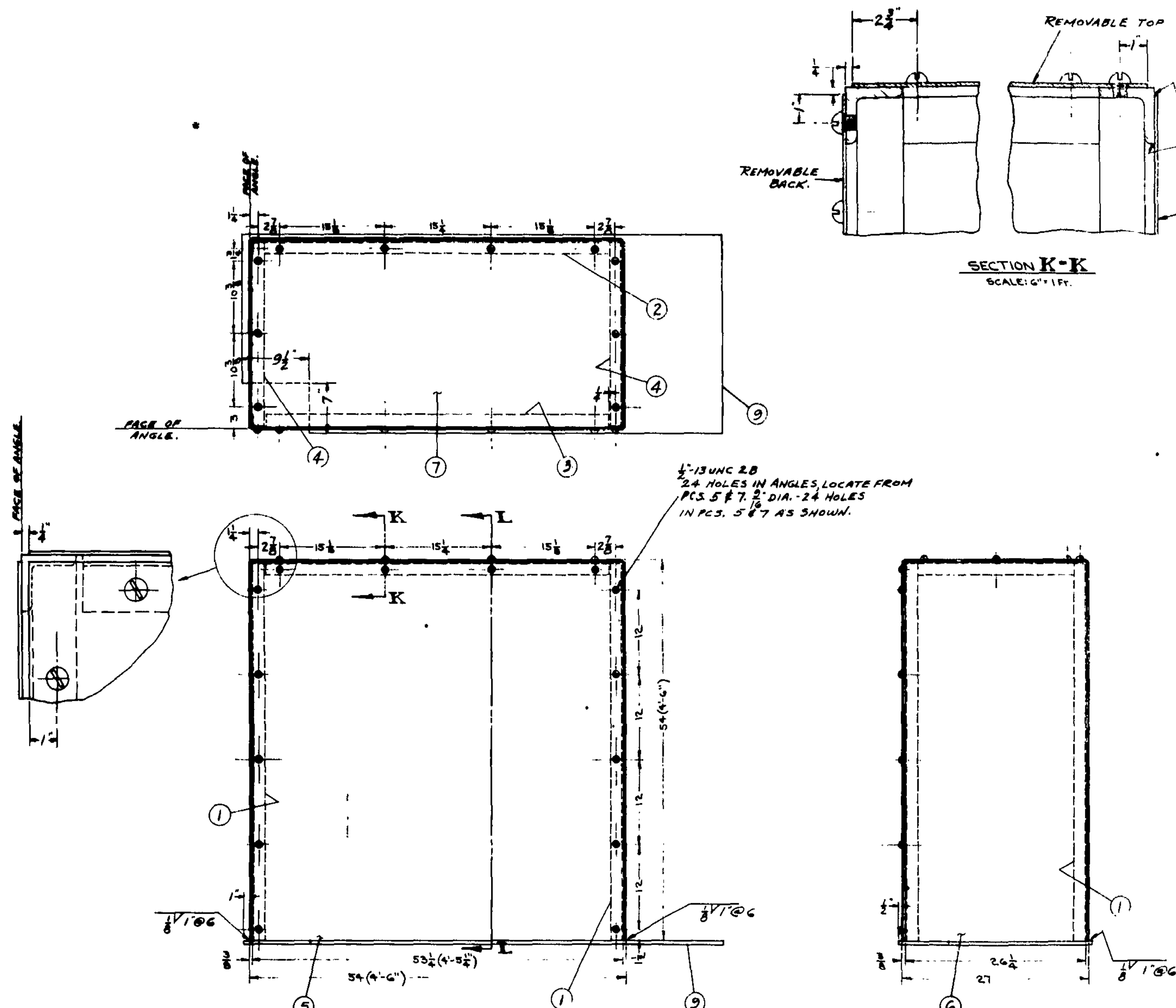
PRIMARY SYSTEM-
AUXILLIARY SKID
DETAILS-I

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-68

SCALE 1 1/2" = 1'-0"

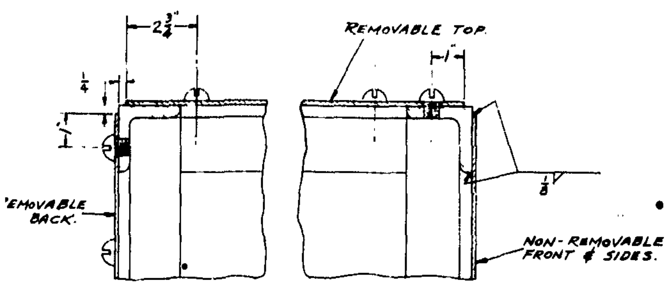
APPLICATION | QTY REQD



ALL MATERIAL TO BE HIGH STRENGTH LOW ALLOY STEEL A.S.T.M. SPEC. A242.

1

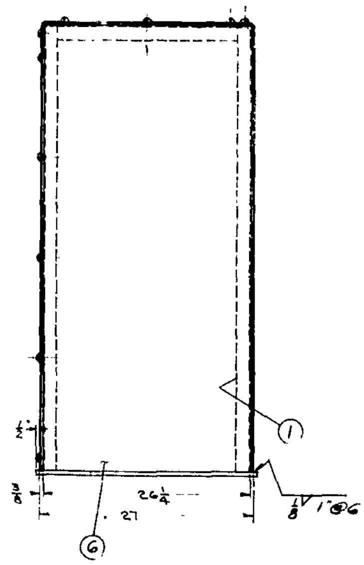
A	ADDED DKG NO. M11594-69 ECO 0292 REF. DWG. ADDED		

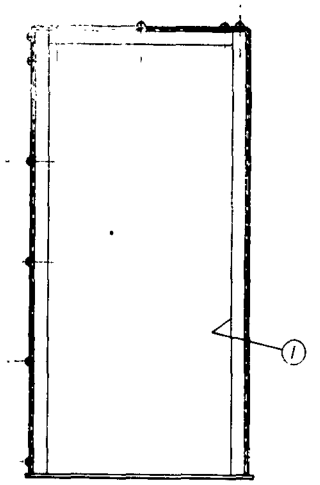


SECTION K-K
SCALE: 6" = 1 FT.

9

IN ANGLES, LOCATE FROM
$\frac{1}{2}$" DIA. - 24 HOLES
7 AS SHOWN.





SECTION L-L

2

M11594-69
A

REF DWG. M11594-68

ALL MATERIAL TO BE HIGH STRENGTH LOW ALLOY STEEL A.S.T.M. SPEC. A 242.

PC. No.	QUAN.	DESCRIPTION	WEIGHT	MATERIAL	SPECIFICATION
9	1	$\frac{1}{2}$ x 28" PLATE - 69" LG. (5'-9")	273.5#	HIGH STRENGTH STL.	ASTM - A-242
8	24	$\frac{1}{2}$"-13 UNC x $\frac{3}{4}$" LG. RD. HD. CAP SCR.	1.7#	STEEL	ASTM A-307
7	1	$\frac{1}{8}$ x 26$\frac{1}{4}$ PLATE - 53$\frac{1}{4}$" LG.	49.5#	HIGH STRENGTH STL.	ASTM - A-242
6	2	$\frac{1}{8}$ x 26$\frac{1}{4}$ PLATE - 53$\frac{1}{4}$" LG.	100.24#	HIGH STRENGTH STL.	ASTM - A-242
5	2	$\frac{1}{4}$ x 53$\frac{1}{4}$" PLATE - 53$\frac{1}{4}$" LG.	202#	HIGH STRENGTH STL.	ASTM - A-242
4	2	2 x 2 x $\frac{1}{4}$ L - 26$\frac{1}{4}$" LG	21.0#	HIGH STRENGTH STL.	ASTM - A-242
3	1	2 x 2 x $\frac{1}{4}$" L - 49$\frac{1}{2}$" LG. (4'-1$\frac{1}{2}$")	19.4#	HIGH STRENGTH STL.	ASTM - A-242
2	1	2 x 2 x $\frac{1}{4}$" L - 49$\frac{1}{4}$" LG. (4'-1$\frac{1}{4}$")	19.5#	HIGH STRENGTH STL.	ASTM - A-242
1	4	2 x 2 x $\frac{1}{4}$ L - 53$\frac{1}{4}$" LG. (4'-5$\frac{1}{4}$")	84.4#	HIGH STRENGTH STL.	ASTM A-242

LIST OF MATERIAL

APPLICATION
QTY REQD

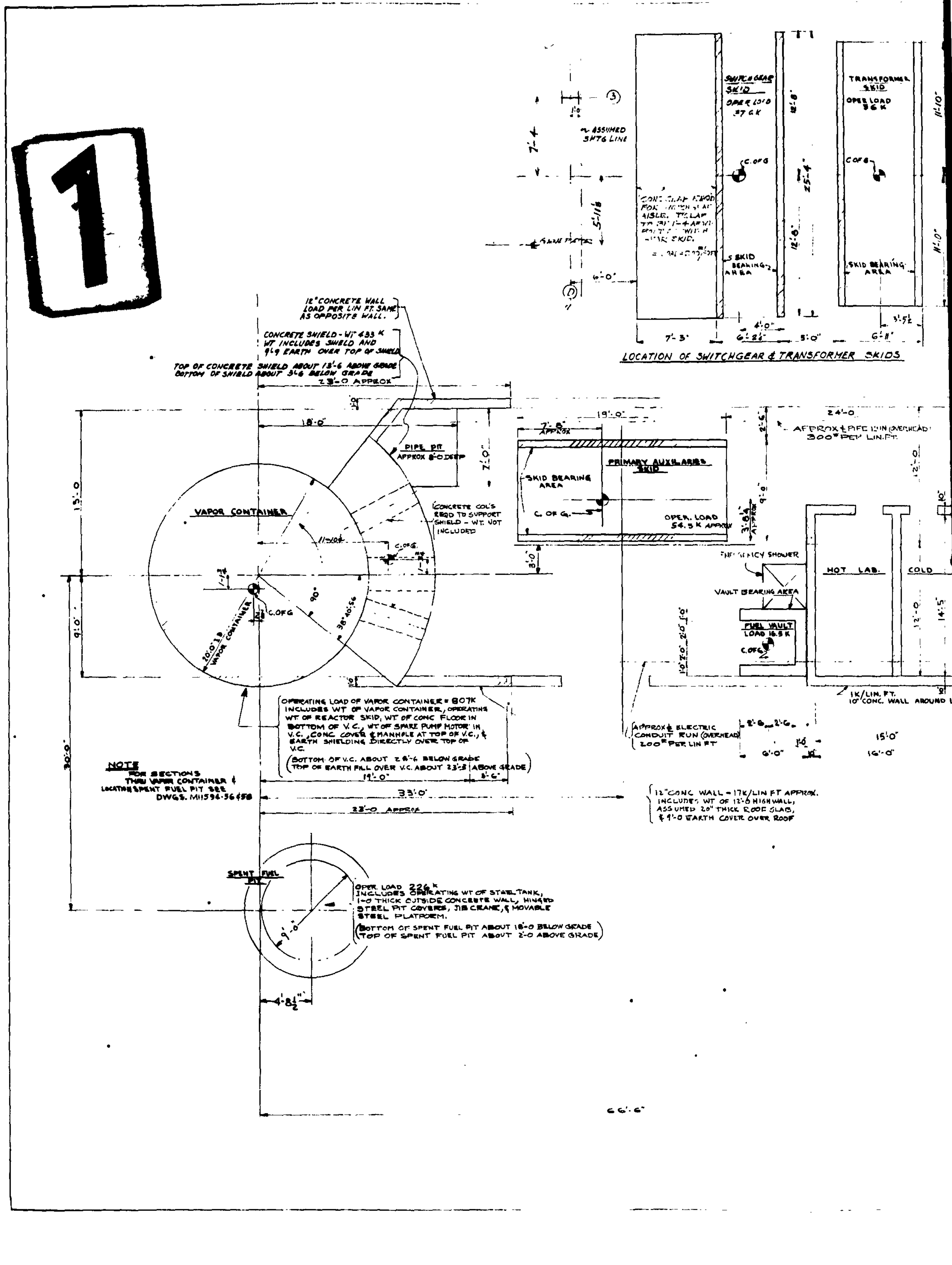
R LAPIVHPON

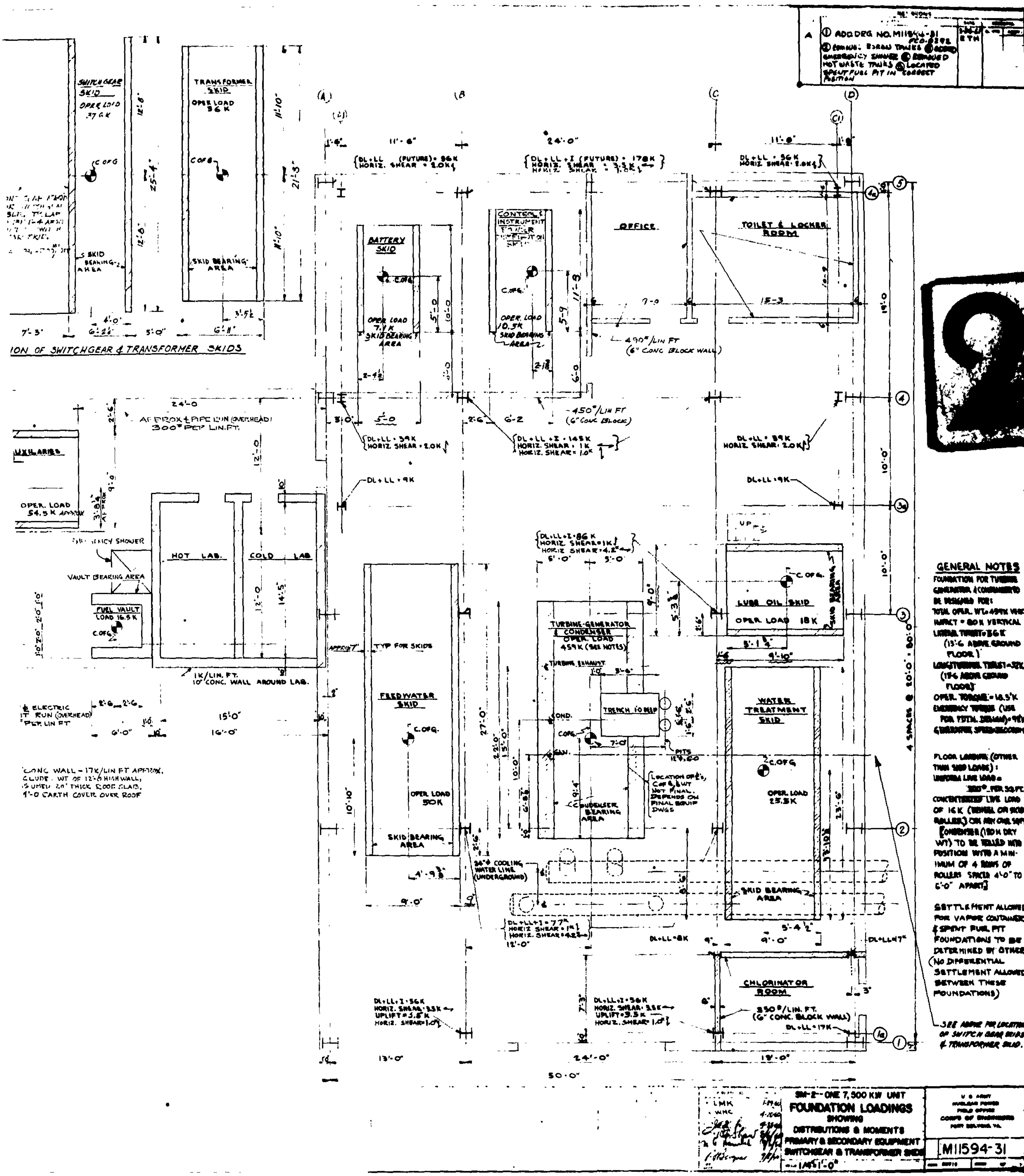
PRIMARY SYSTEM - AUXILLIARY SKID

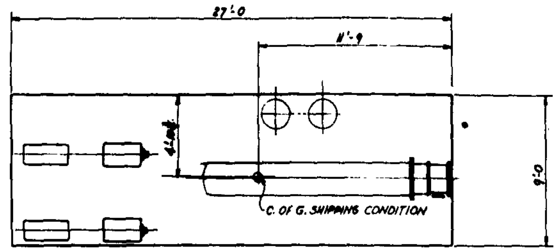
DETAILS-II

M11594-69

SCALE $1\frac{1}{2}$" = 12"

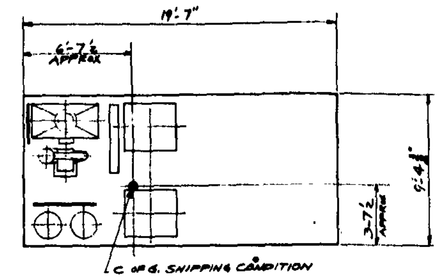

1
LOCATION OF SWITCHGEAR & TRANSFORMER SKIDS
SWITCHGEAR SKID
OPER LOAD 37.6 K
TRANSFORMER SKID
OPER LOAD 36 K
SKID BEARING AREA
ASSUMED SHTG LINE
12" CONCRETE WALL LOAD PER LIN FT SAME AS OPPOSITE WALL.
CONCRETE SHIELD - WT 433 K WT INCLUDES SHIELD AND 9'-9 EARTH OVER TOP OF SHIELD
TOP OF CONCRETE SHIELD ABOUT 13'-6 ABOVE GRADE
BOTTOM OF SHIELD ABOUT 3'-6 BELOW GRADE
23'-0 APPROX
PIPE PIT APPROX 8'-0 DEEP
VAPOR CONTAINER
CONCRETE COL'S REQD TO SUPPORT SHIELD - WT. NOT INCLUDED
C. OF G.
PRIMARY AUXILIARIES SKID
OPER. LOAD 54.5 K APPROX
APPROX ⊄ PIPE RUN (OVERHEAD) 300# PER LIN FT
EMERGENCY SHOWER
VAULT BEARING AREA
FUEL VAULT LOAD 16.5 K
HOT LAB.
COLD
1K/LIN. FT. 10" CONC. WALL AROUND
OPERATING LOAD OF VAPOR CONTAINER = 807K INCLUDES WT OF VAPOR CONTAINER, OPERATING WT OF REACTOR SKID, WT OF CONC FLOOR IN BOTTOM OF V.C., WT OF SPARE PUMP MOTOR IN V.C., CONC. COVER & MANHOLE AT TOP OF V.C., & EARTH SHIELDING DIRECTLY OVER TOP OF V.C.
(BOTTOM OF V.C. ABOUT 28'-6 BELOW GRADE TOP OF EARTH FILL OVER V.C. ABOUT 23'-3 ABOVE GRADE)
APPROX ⊄ ELECTRIC CONDUIT RUN (OVERHEAD) 200# PER LIN FT
12" CONC WALL = 17K/LIN FT APPROX. INCLUDES WT OF 12'-6 HIGH WALL, ASSUMED 20" THICK ROOF SLAB, & 9'-0 EARTH COVER OVER ROOF
NOTE
FOR SECTIONS THRU VAPOR CONTAINER & LOCATING SPENT FUEL PIT SEE DWGS. M11594-56458
33'-0
SPENT FUEL PIT
OPER LOAD 226 K INCLUDES OPERATING WT OF STEEL TANK, 1-0 THICK OUTSIDE CONCRETE WALL, HINGED STEEL PIT COVERS, JIB CRANE, & MOVABLE STEEL PLATFORM.
(BOTTOM OF SPENT FUEL PIT ABOUT 18'-0 BELOW GRADE TOP OF SPENT FUEL PIT ABOUT 2'-0 ABOVE GRADE)
66'-6"


SM-2 – ONE 7,500 KW UNIT
FOUNDATION LOADINGS
SHOWING
DISTRIBUTIONS & MOMENTS
PRIMARY & SECONDARY EQUIPMENT
SWITCHGEAR & TRANSFORMER SKIDS
M11594-31
LMK
WHC
GENERAL NOTES
FOUNDATION FOR TURBINE GENERATOR & CONDENSER TO BE DESIGNED FOR:
OPER. TORQUE = 18.5'K
FLOOR LOADINGS (OTHER THAN SKID LOADS):
CONCENTRATED LIVE LOAD OF 16K
CONDENSER (120K DRY WT) TO BE ROLLED INTO POSITION WITH A MINIMUM OF 4 ROWS OF ROLLERS SPACED 4'-0" TO 6'-0" APART
SETTLEMENT ALLOWED FOR VAPOR CONTAINER & SPENT FUEL PIT FOUNDATIONS TO BE DETERMINED BY OTHERS (NO DIFFERENTIAL SETTLEMENT ALLOWED BETWEEN THESE FOUNDATIONS)
SEE ABOVE FOR LOCATION OF SWITCH GEAR SKIDS & TRANSFORMER SKID.
ADD DRG NO. M11594-31
ION OF SWITCHGEAR & TRANSFORMER SKIDS
SWITCHGEAR SKID
TRANSFORMER SKID
OPER LOAD 36 K
SKID BEARING AREA
BATTERY SKID
OPER LOAD 7.1 K
CONTROL & INSTRUMENT
OPER. LOAD 10.5K
OFFICE
TOILET & LOCKER ROOM
490#/LIN FT (6" CONC BLOCK WALL)
450#/LIN FT (6" CONC BLOCK)
APPROX ¢ PIPE RUN (OVERHEAD) 300# PER LIN. FT.
DL+LL = 39K HORIZ. SHEAR = 2.0K
DL+LL+I = 145K
DL+LL = 9K
OPER. LOAD 54.5 K
HOT LAB.
COLD LAB
VAULT BEARING AREA
FUEL VAULT LOAD 16.5 K
1K/LIN. FT. 10" CONC. WALL AROUND LAB.
LUBE OIL SKID
OPER LOAD 18K
TURBINE-GENERATOR & CONDENSER OPER. LOAD 459K (SEE NOTES)
TYP FOR SKIDS
FEEDWATER SKID
OPER LOAD 50K
TRENCH IO HIP
CONDENSER BEARING AREA
WATER TREATMENT SKID
OPER. LOAD 23.5K
24"¢ COOLING WATER LINE (UNDERGROUND)
CHLORINATOR ROOM
350#/LIN. FT. (6" CONC. BLOCK WALL)
DL+LL = 17K
DL+LL = 8K
DL+LL+I = 56K
4 SPACES @ 20'-0" = 80'-0"
50'-0"
24'-0"
13'-0"
18'-0"



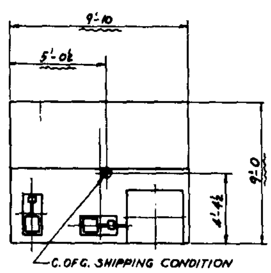
FEEDWATER SKID

SHIPPING WEIGHT-36200#



PRIMARY AUXILIARY SKID

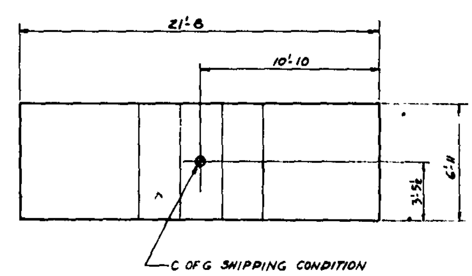
SHIPPING WEIGHT-39,700#(APPROX)



L.O. STORAGE & PURIFIER SKID

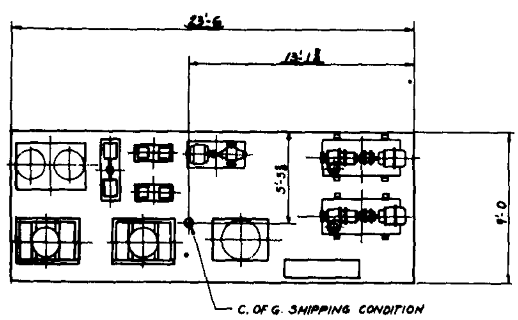
SHIPPING WEIGHT-7,700#

1



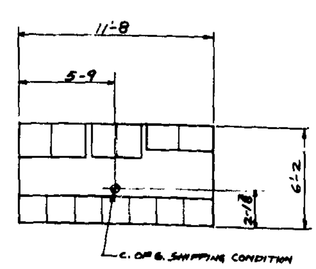
TRANSFORMER SKID

SHIPPING WEIGHT-36000#



WATER TREATMENT SKID

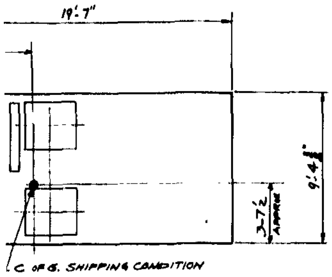
SHIPPING WEIGHT-18,100#



CONTROL & INSTRUMENT POWER DISTRIBUTION SKID

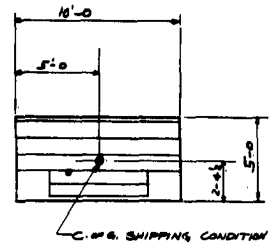
SHIPPING WEIGHT-10500#

REV	DESCRIPTION	DATE
A	① ADD DWG. NO. M11594-36 ECO ② PRIM AUX SKID 9'-4" DIM WAS 5'-0" 0292	3-?-61 ESK



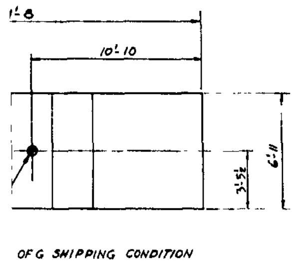
Y AUXILIARY SKID

'PING WEIGHT • 39,700# (APPROX)



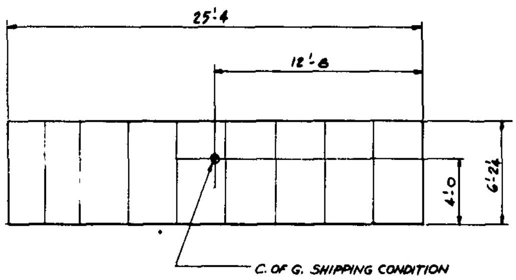
BATTERY SKID

SHIPPING WEIGHT • 7100#



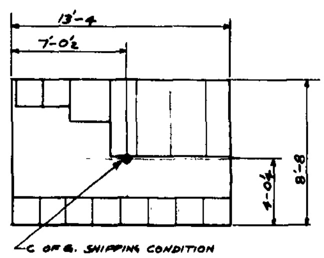
ORMER SKID

NG WEIGHT • 36000#



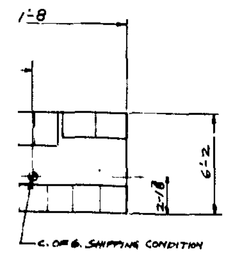
SWITCHGEAR SKID

SHIPPING WEIGHT • 37,600#



MOTOR CONTROL SKID

SHIPPING WEIGHT • 17800#



& INSTRUMENT
STRIBUTION SKID

G WEIGHT • 10500#

KELLMAN 4-6-61	SM-2 — ONE 7,500 K.W. UNIT
WHC 4-11-61	CENTERS OF GRAVITY FOR SKID INSTALLATIONS FOR EQUIPMENT WATER TREATMENT, LUBRICATING OIL & FEEDWATER PRIMARY AUXILLIARY & ELECTRICAL
	SCALE 1/8" = 1'-0"
	U. S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.
	M11594-32

APPLICATION

QTY REQD

8.0 PRIMARY SYSTEM COMPONENTS

8.1 STEAM GENERATOR

8.1.1 Description

The steam generator is a vertically oriented shell and tube heat exchanger mounted on the primary skid inside the vapor container. Steam to run the turbine and the secondary system auxiliaries is produced on the shell side by heat from the higher temperature primary water circulating in the tubes. Design calculations for the steam generator are contained in Volume 2.

The shell side of the steam generator is divided into a boiling section and superheating section separated by a vertical steel baffle plate (Dwg. M 11594-41).

The shell consists of a vertical cylinder of 49 in. dia, 1-1/2 in. thick. An ellipsoidal head is welded to the top of the cylinder and the tube sheet is welded to the bottom of the cylinder. The shell and tube assembly is connected to a channel and can be removed from this channel for shipment or maintenance purposes.

The channel consists of steel hemisphere, stainless steel clad, and a steel flange. It is divided in two compartments by a partition plate. Four-inch inspection openings provide access to each compartment for tube plugging and inspection. The channel has a 14 in. inlet and a 16 in. outlet connection. A 1 in. drain connection will be located in the bottom of the channel. A notch in the partition plate above the drain connection will allow some water flow between the two compartments, thus preventing harmful crud accumulation in the dead corners near the partition plate.

The tube bundle consists of 678 U-bends,of which 606 U-bends are located in the boiling section and 72 U-bends in the superheater section. The tubes are arranged in triangular pitch and are made of type 304 ss. The tube size is 3/4 in. 16 BWG. The 678 U-bends include the tube surface required to permit plugging of 5 percent of the tubes in boiling and superheater section during the lifetime of the steam generator.

In the boiling section of the steam generator, the riser is formed by the space enclosed by a 42 in. ID shroud around the tube bundle and the shell partition plate which separates the boiling section from the superheater section. The downcomer is the annular space between shroud and shell. It is separated from the superheater section by the same partition plate.

The steam-water mixture emerging from the riser during operation leaves the riser through a radial fan-type separator which induces separation of steam and water by centrifugal action. The separated water flows into the downcomer and the steam rises to the inlet of the steam purifier, which is mounted in the top of the boiling section. The purifier, which operates on the principal of centrifugal action, delivers essentially dry steam to the superheater section of the steam generator under normal operating conditions.

The recirculated water mixes with the feedwater in the top of the downcomer. The feedwater inlet consists of a 4 in. diam perforated pipe (torus) around the shroud in the widened portion of the downcomer. This pipe will distribute the colder feedwater evenly over the entire cross section of the downcomer. The presence of the colder feedwater in the top of the downcomer will aid in the prevention of liquid level swell due to steam formulation in the saturated downcomer liquid during increasing load transients.

A 1-in. sparger ring located above the feedwater ring provides for surface blowdown of the steam generator.

A 1-in. syphon type shell drain has been provided for the boiling section and also one for the superheat section of the steam generator. Both drains reach down to the bottom of the shell.

The recirculated water enters the riser in the bottom of the downcomer through port holes in the shroud. Baffle plates are located in the riser to promote uniform flow and temperature distribution. Four 2 in. drains will drain the accumulated water from the purifier. The ends of the drains are located in seal pots in the downcomer. A 16 in. ID manway located in the shell between the two moisture separators provides access to the boiler section of the steam generator. The tube bundle of the superheater section is enclosed by a shroud, by the partition plate between superheater and boiling section and by a plate in the top. The steam eneters through an opening in the top part of the shroud. Baffle plates within the superheater promote uniform flow and temperature distribution. The steam passes through the tube bundle and escapes through an opening in the bottom of the shroud to the annular space between shroud and shell and from there to the 10 in. steam outlet.

8.1.2 Steam Generator Data

Service of unit: Steam generator with superheater
Surface per unit, ft^2: 2905 (3186 gross)

Performance (Overall)

	Shell Side	Tube Side
Fluid	Water, steam	Primary water
Total fluid entering, lb/hr	99,372	7800 gpm
liquid, lb/hr	99,372	3,050,000
Fluid vaporized, lb/hr	99,062	
Blowdown, lb/hr	310	

Performance (Overall) Cont'd

	Shell Side	Tube Side
Specific Heat liquid, Btu/lb/°F		1.181
Temperature in, °F	339.3	527.7
Temperature out, °F	486.2	502.6
Operating pressure, psia	480 (outlet)	2000
Number of passes per shell	1	2
Pressure drop, psi		12.0*
Fouling resistance	.0003 (overall)	
Heat exchanged, Btu/hr		90,495,000

Construction

Design pressure, psia	950	2200
Test pressure, psia	1425	3300
Design temperature, °F	600	600

Tubes

Material	SA-213 type 304
Number/O.D./BWG	678 U's/3/4"/16 BWG
Pitch, (in.)	1
Arrangement	Triangular

Shell

Material	SA-312-B
ID (in.)	49
Thickness, (in.)	1.5
Shell cover material	SA-212-B
Channel material	SA-212-B, SS clad type 304
Tube sheet material	SA-105II, SS overlay, type 304
Connections	
Shell in, (in.)	4
Shell out, (in.)	10
Channel in, (in.)	14
Channel out, (in.)	16

Performance Data (Boiling Section)

Total fluid entering, lbs/hr	99,372	2,766,000
liquid	99,372	2,766,000
Fluid vaporized	99,062	
Blowdown	310	

* Based on 5 percent of tubes plugged

	Shell Side	Tube Side
Spec. heat liquid		1.181
Temperature in	339.3	527.7
Temperature out	464.5	500.6
Operating pressure, psia	488 (avg.)	2000
Number of passes per shell		2
Velocity, ft/sec		13.1
Pressure drop, psi		12.0
Fouling resistance	.0003 (overall)	
Surface, ft^2	2518 (2750 gross)	
Heat exchanged, Btu/hr	88,600,000	
MTD (corrected), ^oF	48.4	
Transfer rate (service), Btu / Hr, Ft^2, ^oF	728	

Construction (Boiling Section)

Tubes	
Material	SA-212, TP 304
Number/OD/BWG	606 U's 3/4 in. 16 BWG
Average length, (in.)	126 straight

Performance Data (Superheat Section)

Fluid	Steam	Primary water
Total fluid entering, lbs/hr	99,062	283,600
Vapor	99,062	
Liquid		283,600
Temperature in, ^oF	464.5	527.7
Temperature out, ^oF	486.2	522.1
Operating pressure, psia	480 (outlet)	2000
Number of passes		2
Velocity, ft/sec		11.3
Pressure drop, psi		12.0*
Fouling resistance	.0003 (overall)	
Surface, ft^2	387 (436 gross)	
Heat exchanged, Btu/hr	1,891,000	
MTD (corrected), ^oF	49.1	
Transfer (service)	99.5	

Construction (Superheat Section)

Tubes	
Material	SA-212, Tp 304
Number/OD/BWG	72 U's, 3/4 in. 16 BWG
Average length, (in.)	168 (straight)

* Based on 5 percent of tubes plugged

8.2 PRESSURIZER

The pressurizer has two basic functions. During normal operating conditions, it maintains pressure over the entire primary circulating system. This is accomplished by keeping the contained water at the saturated temperature, corresponding to the primary system pressure.

The pressurizer also suppresses pressure excursions during transients. The pressurizer volume contains 38.8 ft^3 of steam and 13,85 ft^3 of water and is sufficiently large to prevent excessive pressure fluctuations. Design is such that water level will never be less than 3 in. over the heaters. Instantaneous change in rated load from 100 percent to 1 percent results in pressure rise of 150 psi, while load increase from 5 percent to 100 percent in 60 sec results in pressure decrease of 125 psi. For additional details on transient behavior of the pressurizer, refer to APAE Memo 266.[1]

The pressurizer (Dwg. M 11594-40) is a carbon steel, vertically mounted cylindrical vessel closed with two SC hemispherical heads. All surfaces in contact with the primary water (vapor) are overlayed with stainless steel (3/16 in. thick min). Design (operating) pressures and temperatures are 2200 psig and 650°F (2000 psig, 636°F). A 16 in. diam manhole is provided at the top of the vessel to permit any repairs on the vessel interiors (heater tubes, and the like). The top hemisphere also contains a 3 in. - 2500 lb N.N. flanged connection for one safety valve assembly consisting of two 1-1/2 x 2-1/2 in. safety valves. The number of safety valves is in accordance with Code requirements (Case 1271-N).

Two thermocouples located 90° apart complete the number of penetrations located in the upper hemisphere. The cylindrical portion of the vessel consists of an upper section, containing a 2 in. cooling nozzle connected to the 14 in. primary piping (on pump discharge side), and a lower section, containing 26 in. heaters, each rated at 1-1/2 KW. This section is thick enough to provide the necessary reinforcement for heater penetrations. Heater units are of bayonet type, enclosed by stainless steel tubes (.570 ID x .880 OD), with replaceable cartridges. They are arranged in two horizontal planes 3 in. apart with heaters in each plane (Section B-B on Dwg. M 11594-40).

Power (480 V) to the electrical heaters is supplied through two 1-1/4 in. penetrations located in each of 2 terminal boxes. Heaters are wired in 3 banks, two with 9 heaters each and 1 bank with 8 heaters. During normal operating conditions, only a fraction of the total heater capacity installed is needed. This is accomplished by voltage regulation.

Heaters are supported by a 3/8 in. stainless steel baffle, located on pressurizer and welded to the vessel through a flexible member (Section BB on Dwg. M 11594-40). This became necessary in view of the differential expansion between the baffle and reactor vessel.

Four 1 in. connections have been provided for two level controller, with two upper connections located 25 in. above the normal water level. The 3 in. nozzle on the bottom of the pressurizer provides for a 3 in pipe connection with the primary piping. The nozzle is equipped with a perforated baffle, directing the flow of cold insurge water (during transients) deep into the vessel interior. It protects the vessel wall from rapid cooling around the nozzle, thus preventing thermal stress buildup.

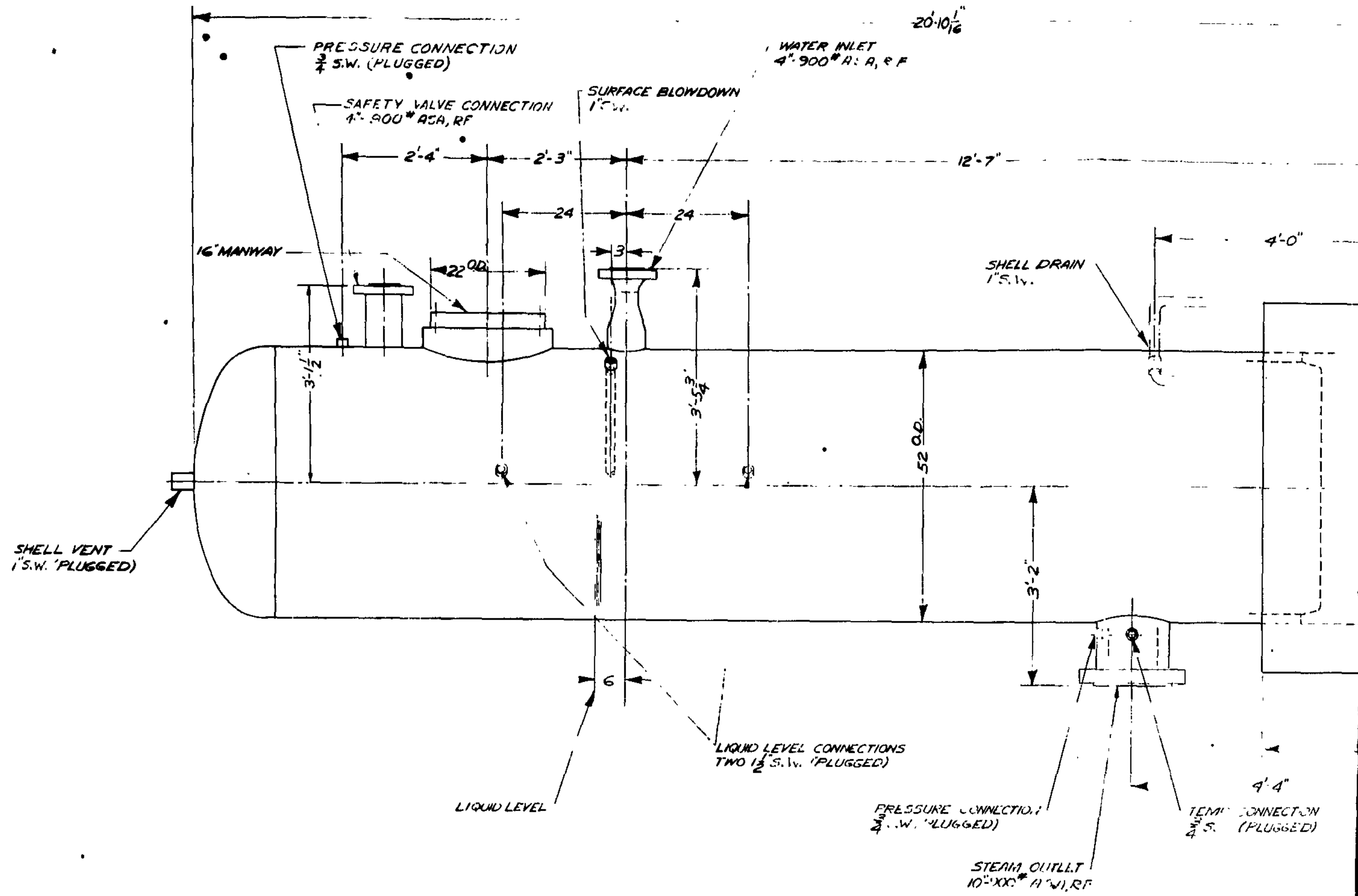
The pressurizer is located outside the primary skid and has its own supporting structure to which it is bolted. Field welding will be required for 3 in. connection with the primary piping. It weighs 10,320 lbs (empty) and has 4 in. thick block insulation (calcium silicate).

8.3 PRIMARY PUMP

The primary coolant pump circulates primary cooling water through the closed primary coolant loop. The pump is a single stage centrifugal unit, and has a vertical shaft, bottom suction, clockwise rotation when looking down on the volute, a horizontal discharge and a canned rotor for zero leakage. The motor, volute-closure and impeller are assembled and removed as a unit. The volute nozzles are butt-welded into the piping loop. The motor assembly is equipped with a heat exchanger capable of keeping the armature operating at a safe temperature. The pump is designed for a 7800 gpm flow with a 114.2 ft IDH and a temperature of 500.5^{o}F, and is driven by a 440 volt, 3 phase, 60 cycle 300 hp motor. The approximate dimensions and general outline of the pump are shown on Dwg M 11594-37. The materials will meet the requirements set forth in ALCO Specification Number ALCO-S-47, dated August 25, 1960.

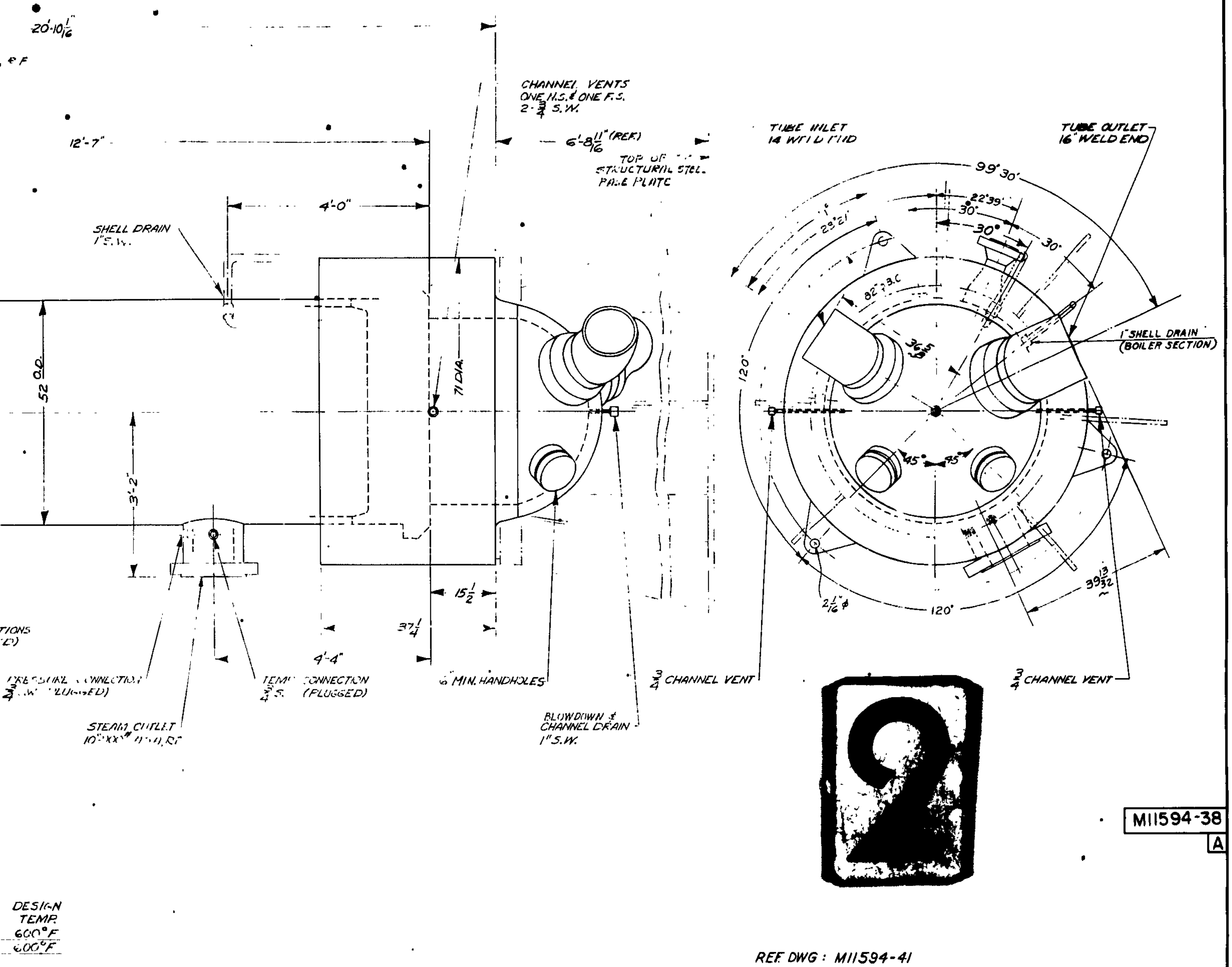
REFERENCES

1. Bradley, P. L., "Development and Design Studies of SM-2 Pressurizer", APAE Memo No. 266, July 29, 1960.



1

	DESIGN PRESS.	TEST PRESS.	DESIGN TEMP.
SHELL	950 PSI	1425 PSI	600°F
TUBES	2300 PSI	3450 PSI	600°F



M11594-38

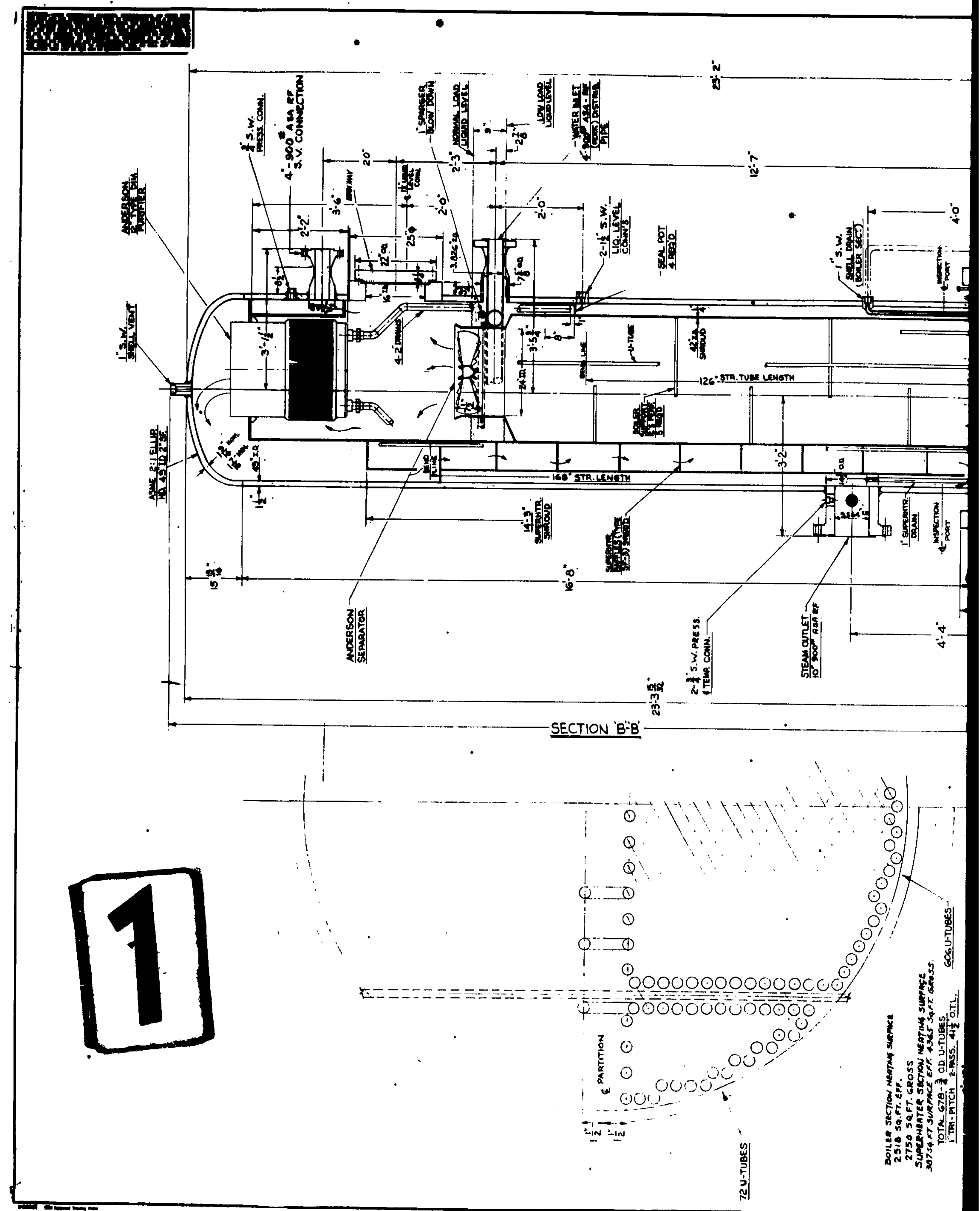
A

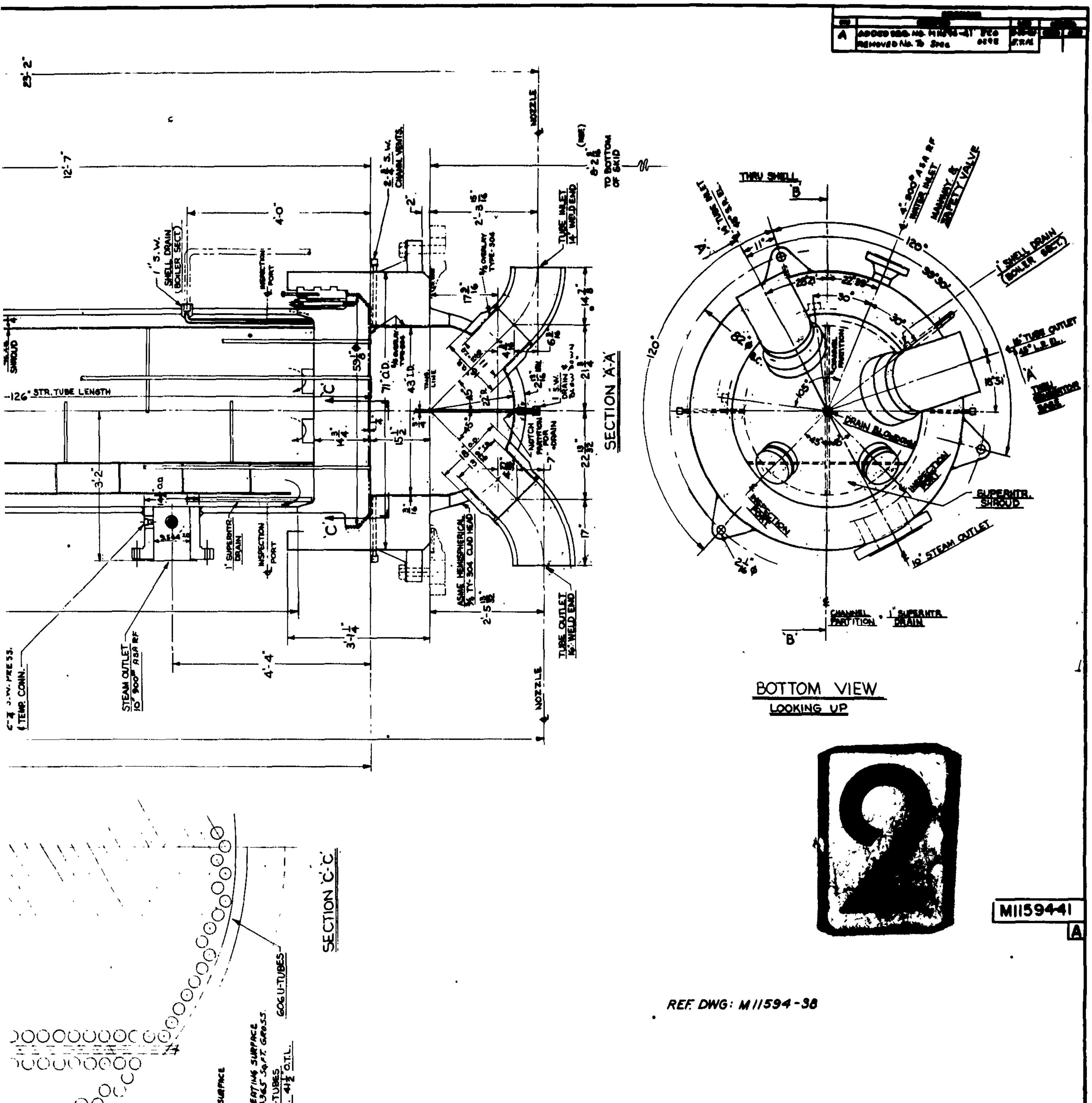
(1)	ADDED DRG. NO. M11594-38	ECO A292	E.T.H.

STEAM GENERATOR OUTLINE

U.S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.

M11594-38


1
SECTION 'B'-'B'
ANDERSON SEPARATOR
ANDERSON 12" TYPE DIA PURIFIER
1" S.W. SHELL VENT
SPARGER BLOW DOWN
NORMAL LOAD LIQUID LEVEL
LOW LOAD LIQUID LEVEL
SEAL POT 4 REQ'D
126" STR. TUBE LENGTH
168" STR. LENGTH
STEAM OUTLET
INSPECTION PORT
SUPERHTR. DRAIN
SUPERHTR. SHROUD
U-TUBE
℄ PARTITION
72 U-TUBES
606 U-TUBES
23'-2"
12'-7"
16'-8"
4'-4"



LIST OF MATERIAL

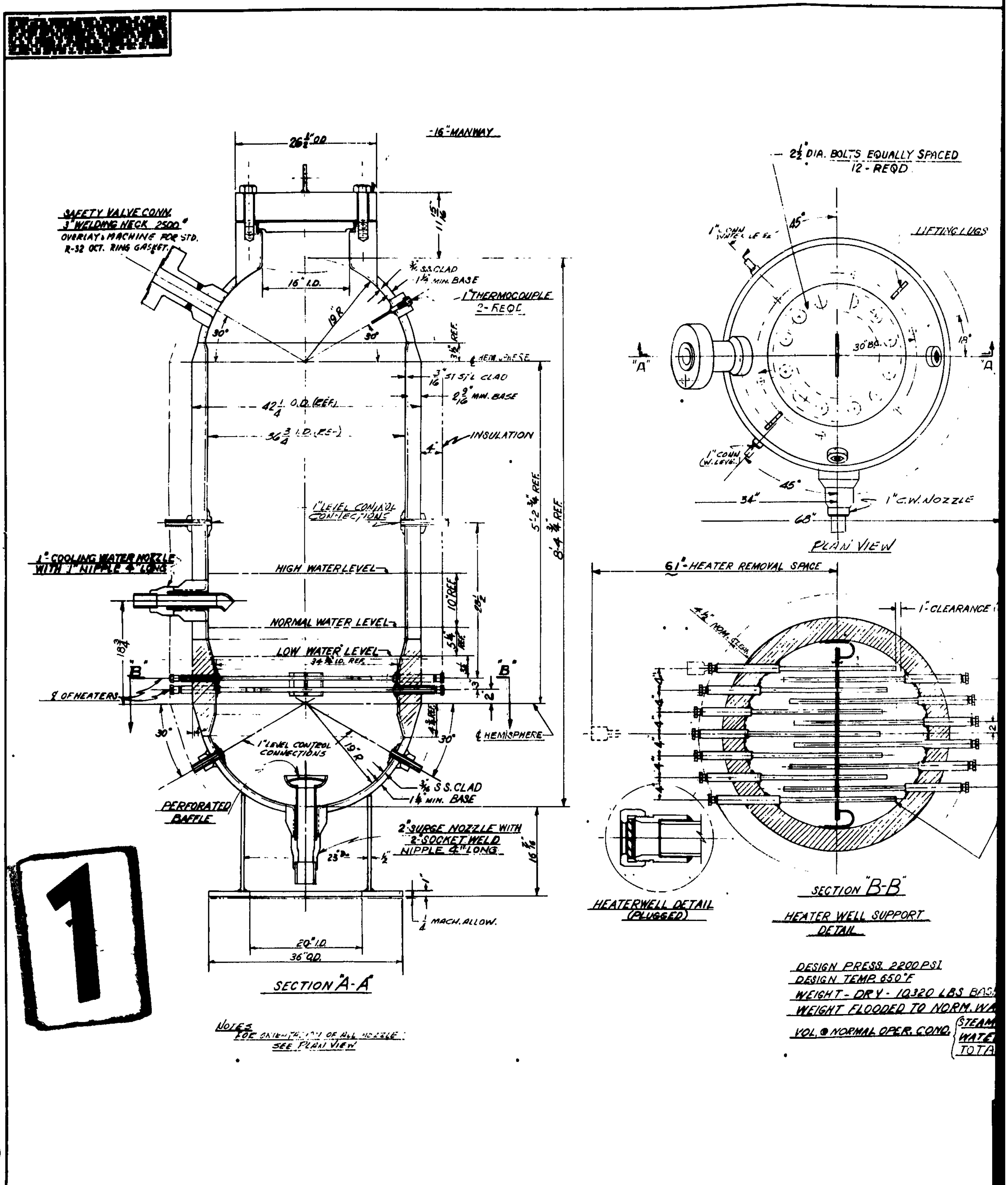
APPLICATION | QTY REQD

STEAM GENERATOR
CROSS SECTION

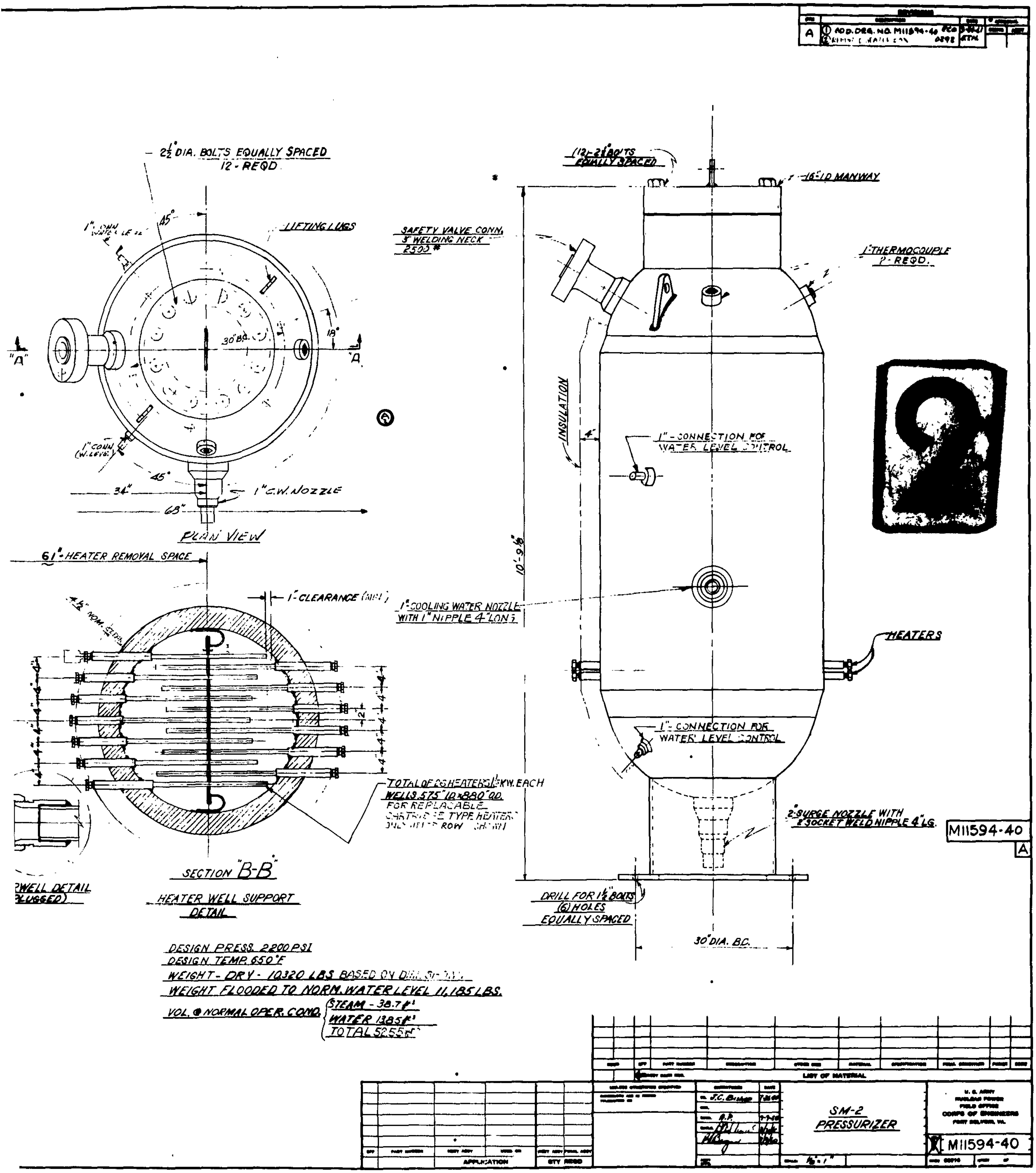
U.S. ARMY
NUCLEAR POWER FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR

SCALE 1" = 1'-0"

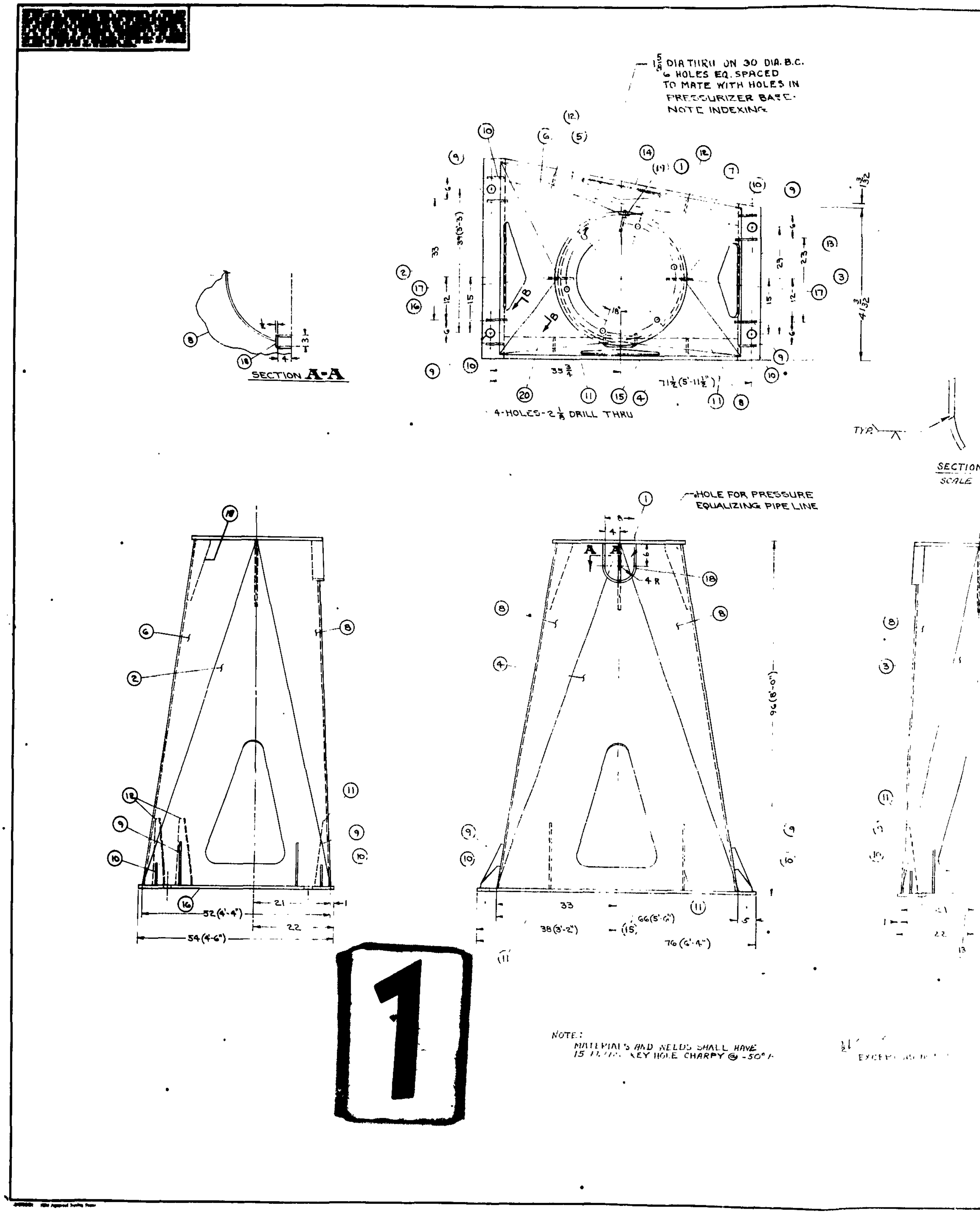
M11594-41



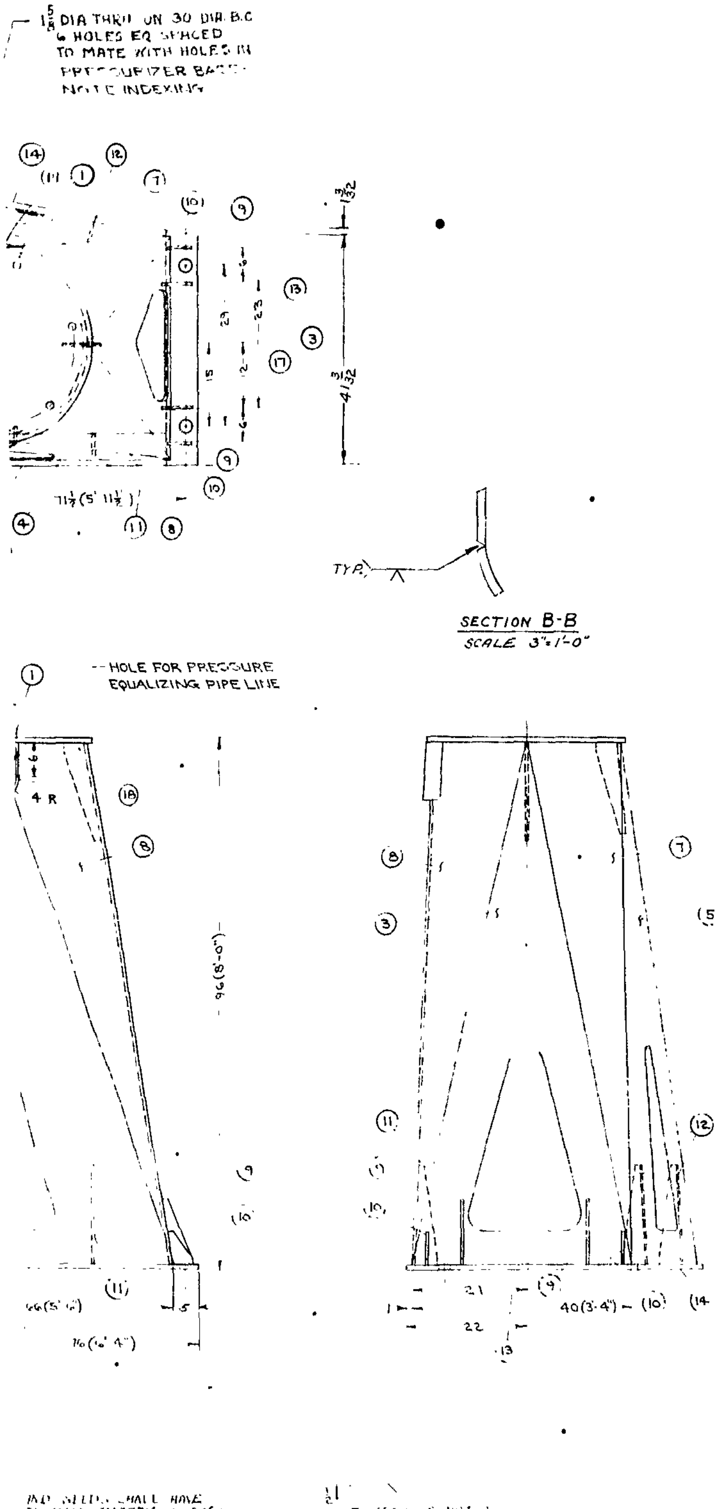
16" MANWAY
26 1/2" O.D.
2 1/2 DIA. BOLTS EQUALLY SPACED
12 - REQD
SAFETY VALVE CONN.
3" WELDING NECK 2500#
OVERLAY & MACHINE FOR STD.
R-32 OCT. RING GASKET
3/16 S.S. CLAD
1 1/4 MIN. BASE
16" I.D.
19 R
1" THERMOCOUPLE
2-REQD
30°
HEMISPHERE
3/16" ST. ST'L CLAD
2 9/16" MIN. BASE
42 1/4 O.D. (REF.)
36 3/4 I.D. (REF.)
INSULATION
4"
5'-2 3/4" REF.
8'-4 3/4" REF.
1" LEVEL CONTROL CONNECTIONS
1" COOLING WATER NOZZLE
WITH 1" NIPPLE 4" LONG
HIGH WATER LEVEL
NORMAL WATER LEVEL
LOW WATER LEVEL
34 1/2" I.D. REF.
10" REF.
29 1/2
5 1/4" REF.
18 3/4
"B"
℄ OF HEATERS
℄ HEMISPHERE
1" LEVEL CONTROL CONNECTIONS
19" R
3/16 S.S. CLAD
1 1/4" MIN. BASE
PERFORATED BAFFLE
2" SURGE NOZZLE WITH
2" SOCKET WELD
NIPPLE 4" LONG
25 3/8"
16 7/8"
1/4" MACH. ALLOW.
20" I.D.
36" O.D.
SECTION "A-A"
NOTES
FOR ORIENTATION OF ALL NOZZLES
SEE PLAN VIEW
1" CONN WATER LEVEL
45°
LIFTING LUGS
30° BC
18"
"A"
1" CONN (W. LEVEL)
34"
68"
1" C.W. NOZZLE
PLAN VIEW
61"- HEATER REMOVAL SPACE
1" CLEARANCE
4 1/2" NOM. CLEAR.
HEATERWELL DETAIL (PLUGGED)
SECTION "B-B"
HEATER WELL SUPPORT DETAIL
DESIGN PRESS. 2200 PSI
DESIGN TEMP. 650°F
WEIGHT - DRY - 10320 LBS
WEIGHT FLOODED TO NORM. WA
VOL. @ NORMAL OPER. COND.
STEAM
WATER
TOTAL
1



A
ADD. DWG. NO. M11594-40
2½" DIA. BOLTS EQUALLY SPACED
12 - REQD.
45°
LIFTING LUGS
30" B.C.
18"
"A"
"A"
1" CONN. (W. LEV.)
45°
34"
1" C.W. NOZZLE
65"
PLAN VIEW
61" - HEATER REMOVAL SPACE
1" CLEARANCE
SECTION "B-B"
HEATER WELL SUPPORT
DETAIL
TOTAL OF 26 HEATERS
WELLS .575" ID x .880" OD.
FOR REPLACABLE
(12) 2¼" BOLTS
EQUALLY SPACED
16" ID MANWAY
SAFETY VALVE CONN.
3" WELDING NECK
2500 #
1" THERMOCOUPLE
2 - REQD.
INSULATION
4"
1" - CONNECTION FOR
WATER LEVEL CONTROL
10'-9⅝"
1" COOLING WATER NOZZLE
WITH 1" NIPPLE 4" LONG
HEATERS
1" - CONNECTION FOR
WATER LEVEL CONTROL
2" SURGE NOZZLE WITH
2" SOCKET WELD NIPPLE 4" LG.
M11594-40
A
DRILL FOR 1½" BOLTS
(6) HOLES
EQUALLY SPACED
30" DIA. B.C.
DESIGN PRESS. 2200 PSI
DESIGN TEMP. 650°F
WEIGHT - DRY - 10320 LBS
WEIGHT FLOODED TO NORM. WATER LEVEL 11,185 LBS.
VOL. @ NORMAL OPER. COND.
STEAM - 38.7 ft³
WATER 13.85 ft³
TOTAL 52.55 ft³
LIST OF MATERIAL
APPLICATION
SM-2
PRESSURIZER
U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.
M11594-40


1 5/8 DIA THRU ON 30 DIA. B.C.
6 HOLES EQ. SPACED
TO MATE WITH HOLES IN
PRESSURIZER BASE.
NOTE INDEXING
SECTION A-A
4-HOLES-2 1/8 DRILL THRU
71 1/2 (5'-11 1/2")
35 3/4
41 3/32
1 3/32
HOLE FOR PRESSURE
EQUALIZING PIPE LINE
4 R
96 (8'-0")
52 (4'-4")
54 (4'-6")
21
22
33
38 (3'-2")
66 (5'-6")
76 (6'-4")
TYP.
SECTION
1
NOTE:
MATERIALS AND WELDS SHALL HAVE

A	ADDED DRG NO. M11594-42 ECO	8-5-61	
	REF. DWGS ADDED	6898	FJH



SECTION B-B
SCALE 3"=1'-0"

REF. DWG'S: M11594-43
M11594-44

M11594-42

20	1		CORNER PL.		M11594-44-12				
19	1		GUSSET		M11594-44-11				
18	1		¼×3×25¾ LG.		STL. PL.	ASTM-A-242			
17	2		GUSSET		M11594-44-10				
16	1		1"×6×54(4'-6") LG.		STL. PL.				
15	1		1"×6×64(5'-4") LG.		STL. PL.				
14	1		PLATE		M11594-44-9				
13	1		PLATE		M11594-44-8				
12	2		GUSSET		M11594-44-7				
11	2		GUSSET		M11594-44-6				
10	4		GUSSET		M11594-44-5				
9	4		GUSSET		M11594-44-4				
8	1		CORNER PL.		M11594-44-3				
7	1		CORNER PL.		M11594-44-2				
6	1		CORNER PL.		M11594-44-1				
5	1		SIDE PL.		M11594-43-5				
4	1		SIDE PL.		M11594-43-4				
3	1		END PL.		M11594-43-3				
2	1		END PL.		M11594-43-2				
1	1		FLANGE		M11594-43-1				

LIST OF MATERIAL

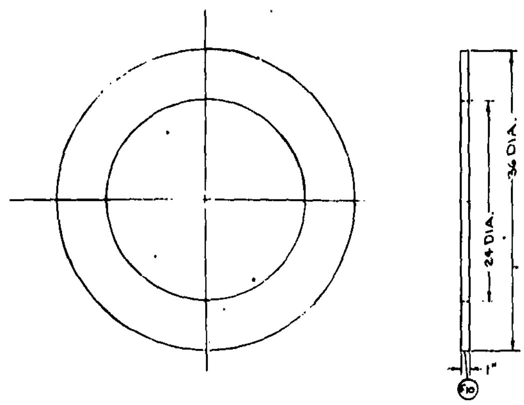
R. LAMPHIRON	7-28-60	PRESSURIZER SUPPORT	U. S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.
Ma. Ar.	9-9-60		
W F S			M11594-42
		SCALE 1" = 1 FT	

APPLICATION

QTY REQD

[redacted]

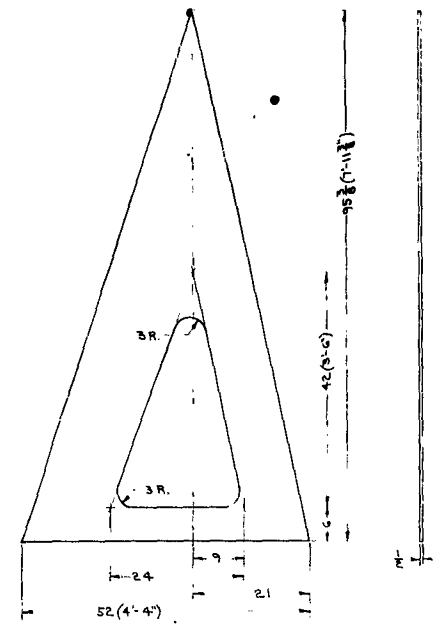
FLANGE – M11594-43-



MATERIAL: STEEL PLATE-ASTM-A 242

SCALE: 1½" = 1 FT.

END PLATE –



F10 ALL CUT EDGES

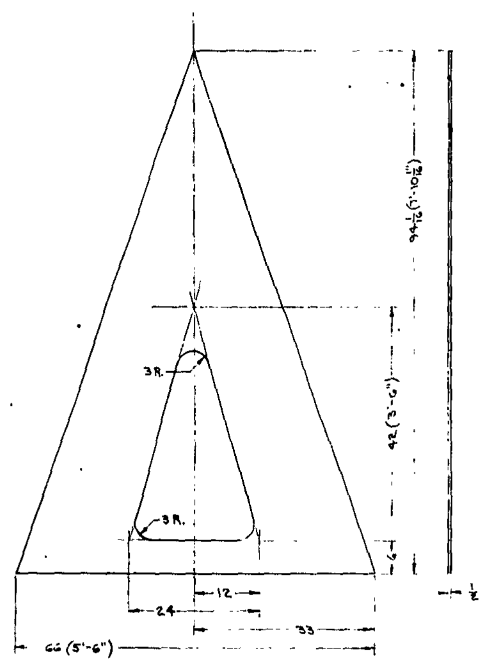
MATERIAL: STEEL PLATE-ASTM-A242

SCALE: 1" = 1 FT.

END

MATER

SIDE PLATE - M11594-43-4

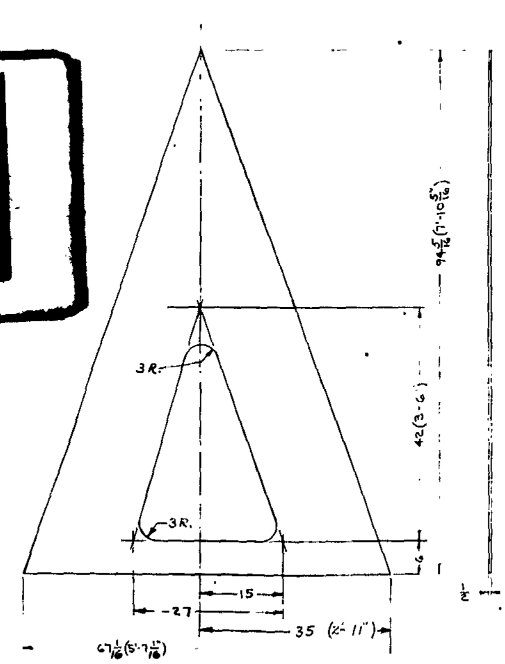


F10 ALL CUT EDGES

MATERIAL: STEEL PLATE-ASTM-A 242

SCALE: 1" = 1 FT.

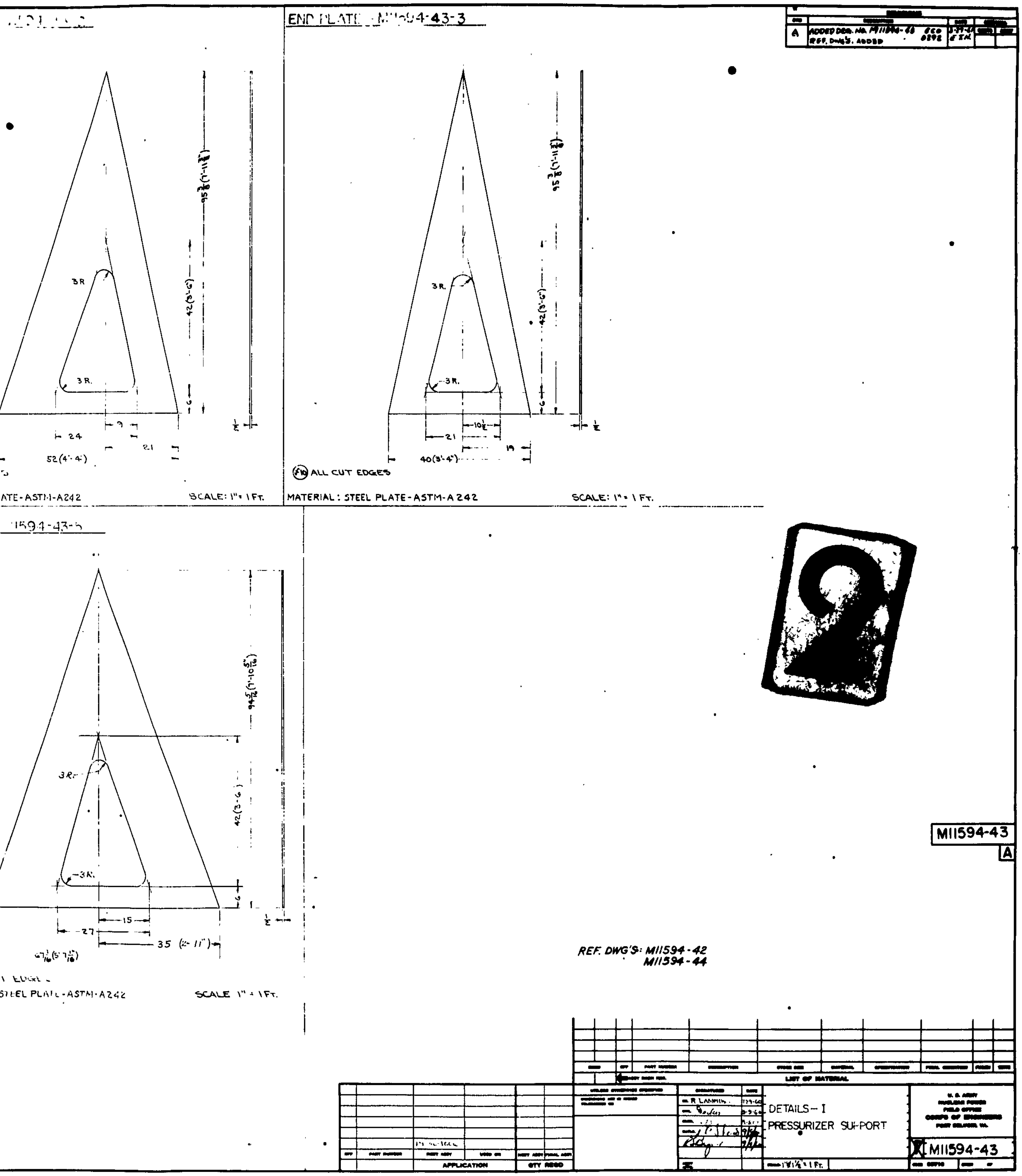
M11594-43-5



F10 ALL CUT EDGES

MATERIAL: STEEL PLATE-ASTM-A 242

SCALE: 1" = 1 FT.

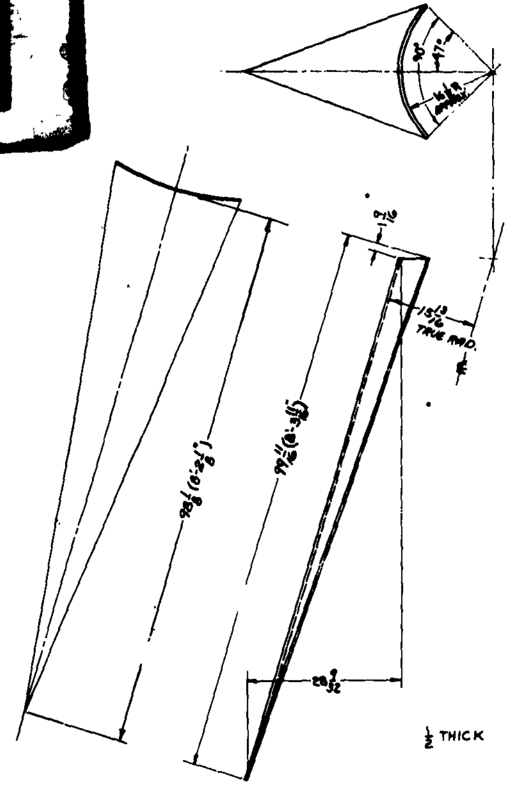


END PLATE M11594-43-3
3R.
3R.
$95\frac{3}{8}$ (7'-11$\frac{3}{8}$")
42 (3'-6")
6
$10\frac{1}{2}$
21
19
40 (3'-4")
ALL CUT EDGES
MATERIAL: STEEL PLATE-ASTM-A242
SCALE: 1" = 1 FT.
24
9
21
52 (4'-4")
ATE-ASTM-A242
SCALE: 1" = 1 FT.
1594-43-5
3R
3R.
$94\frac{5}{16}$ (7'-10$\frac{5}{16}$")
42 (3'-6")
15
27
35 (2'-11")
STEEL PLATE-ASTM-A242
SCALE 1" = 1 FT.
A
ADDED DWG. NO. M11594-68 ECO 0892
REF. DWG'S. ADDED
M11594-43
A
REF. DWG'S: M11594-42
M11594-44
LIST OF MATERIAL
APPLICATION
QTY REQD
R LAMMIS
DETAILS – I
PRESSURIZER SUPPORT
M11594-43

[redacted]

1

CORNER PLATE- M11594-44-1

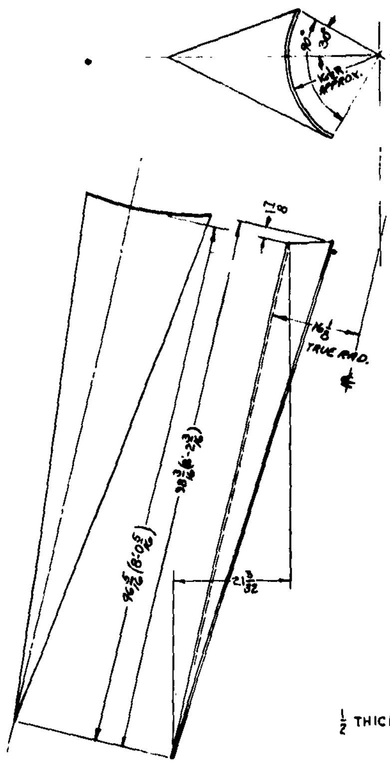


1/2 THICK

MAT'L: STEEL R.- ASTM-A-242 (HIGH STRENGTH, LOW ALLOY)

SCALE: 1" = 1 FT.

CORNER PLATE - M11594-44-2



1/2 THICK

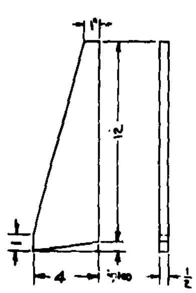
(HIGH STRENGTH, LOW ALLOY)
MAT'L: STEEL R.- ASTM-A-242

SCALE: 1" = 1 FT.

CORNE

(HIGH STR
MAT'L: ST

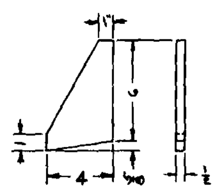
GUSSET- M11594-44-4



(HIGH STRENGTH, LOW ALLOY)
MAT'L: STEEL R.- ASTM-A-242

SCALE: 3" = 1 FT.

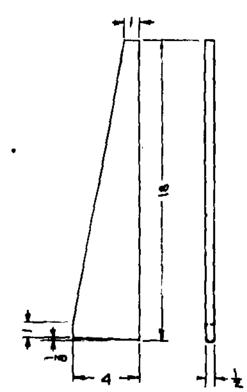
GUSSET- M11594-44-5



(HIGH STRENGTH, LOW ALLOY)
MAT'L: STL.R.-ASTM-A-242

SCALE: 3" = 1 FT.

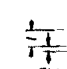
GUSSET- M11594-44-6



MAT'L: STL. R.- ASTM-A-242
(HIGH STRENGTH, LOW ALLOY)

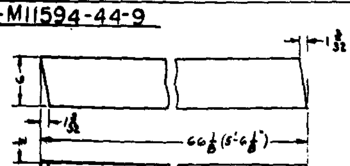
SCALE: 3" = 1 FT.

GUSSET- M 11594



MAT'L: STL R.-ASTM-A-
(HIGH STRENGTH, LOW ALL

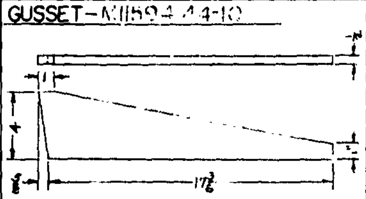
PLATE- M11594-44-9



(HIGH STRENGTH, LOW ALLOY)
MAT'L: STEEL R.- ASTM-A-242

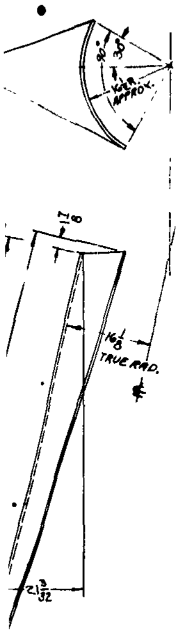
SCALE: 3" = 1 FT.

GUSSET- M11594-44-10

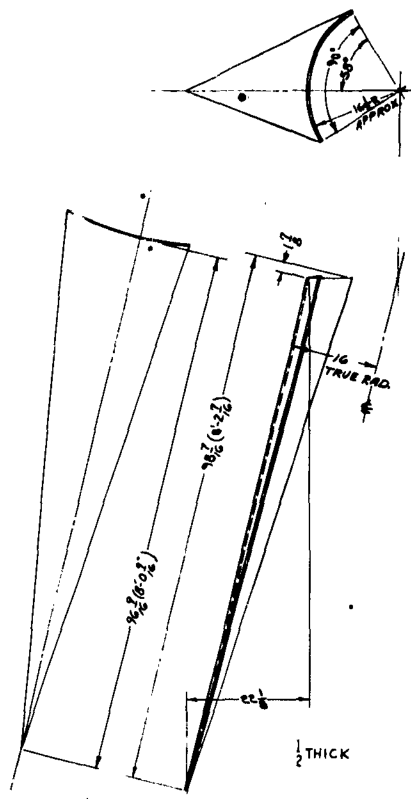


(HIGH STRENGTH, LOW ALLOY)
MAT'L: STEEL R.- ASTM-A-242

SCALE: 3" = 1 FT.

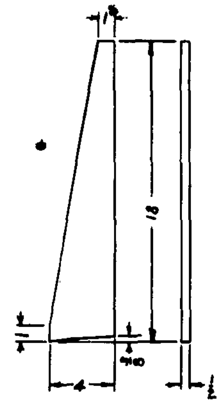


CORNER PLATE - M11594-44-3



(HIGH STRENGTH, LOW ALLOY)
MAT'L: STEEL PL.- ASTM-A-242
SCALE: 1"=1 FT.

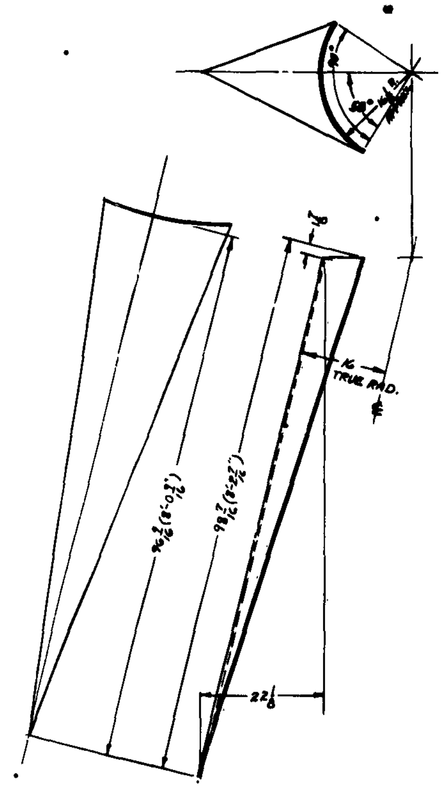
GUSSET- M11594-44-11



MAT'L STL. PL.: ASTM-A-242
(HIGH STRENGTH, LOW ALLOY)
SCALE: 3"=1 FT

REV	DESCRIPTION		
A	ADDED DRG. No. M11594-44 REF. DWG'S. ADDED		

CORNER PLATE - M11594-44-12

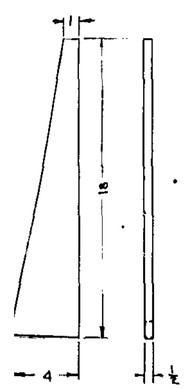


MAT'L: STL. PLATE - ASTM-A-242
(HIGH STRENGTH, LOW ALLOY)
SCALE: 1"=1 FT.

*REF. DWG'S: M11594-42
M11594-43*

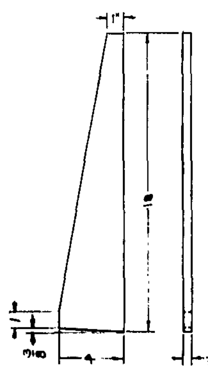
M11594-44
A

-44-6



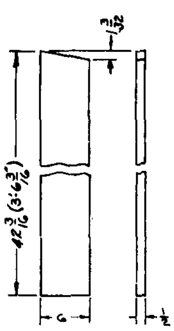
-A 242
(LOY)
SCALE: 3"=1 FT

GUSSET- M11594-44-7



MAT'L: STL. PL.-ASTM-A-242
(HIGH STRENGTH, LOW ALLOY)
SCALE: 3"=1 FT.

PLATE- M11594-44-8



MAT'L:
STL. PL.-ASTM-A-242
(HIGH STRENGTH, LOW ALLOY)
SCALE: $1\frac{1}{2}$"=1 FT.

LIST OF MATERIAL

APPLICATION

QTY REQD

R. LAMPHMAN

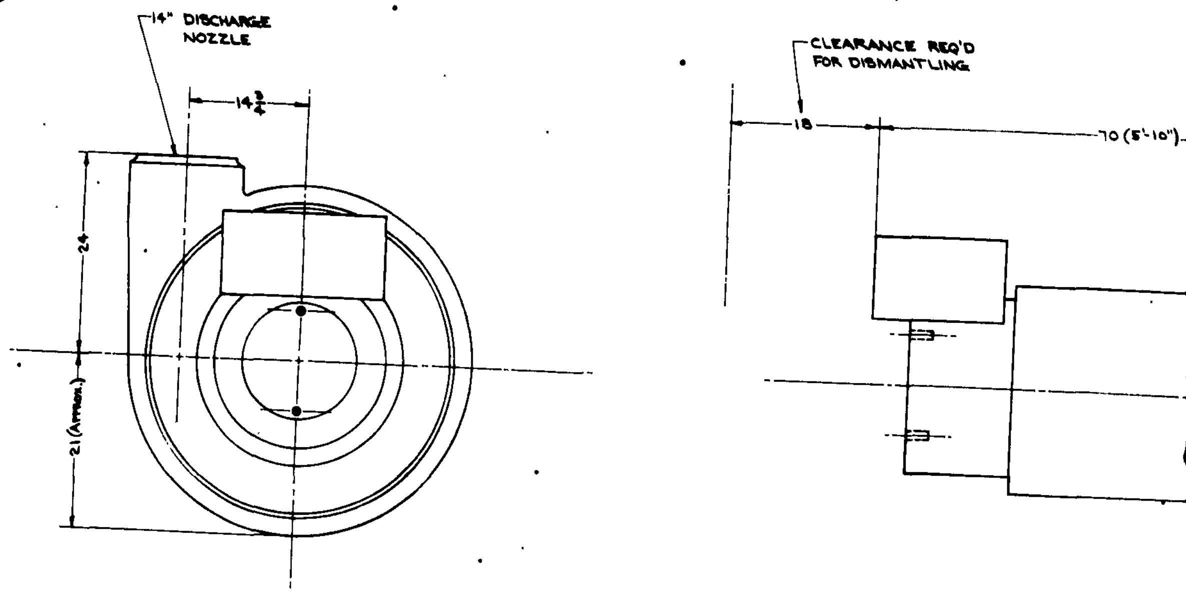
WFS

DETAILS - II
PRESSURIZER SUPPORT

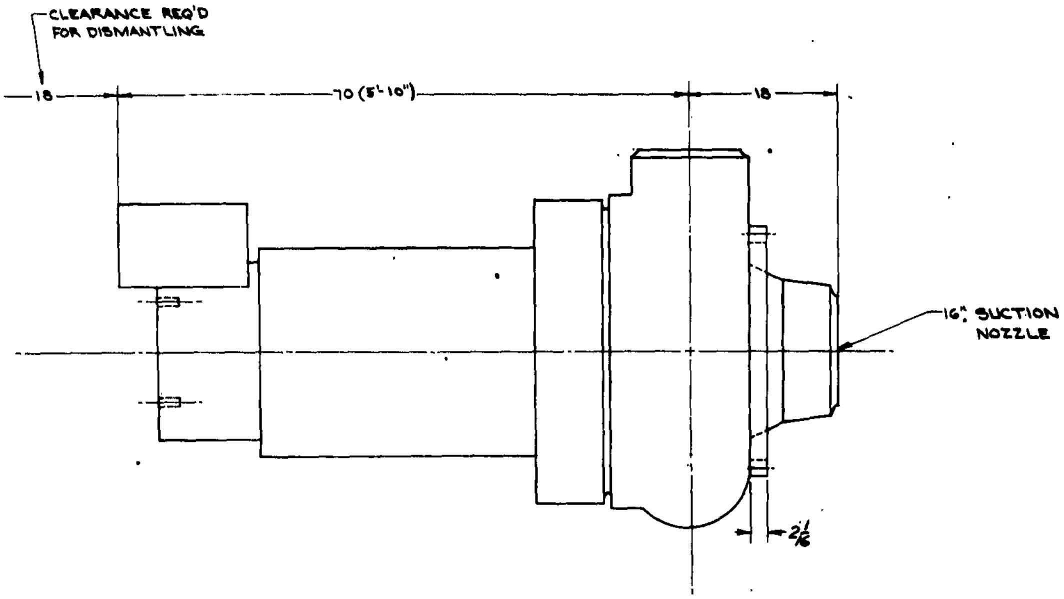
U.S. ARMY
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-44

1


14" DISCHARGE NOZZLE
14 3/4
24
21 (APPROX.)
CLEARANCE REQ'D FOR DISMANTLING
18
70 (5'-10")

A	ADDED DRG. NO. M11594-37		E.T.M.



REFERENCE DWG.
M11594-37 PRIM. SYST. PIPING LAYOUT.

M11594-37
A

LIST OF MATERIAL

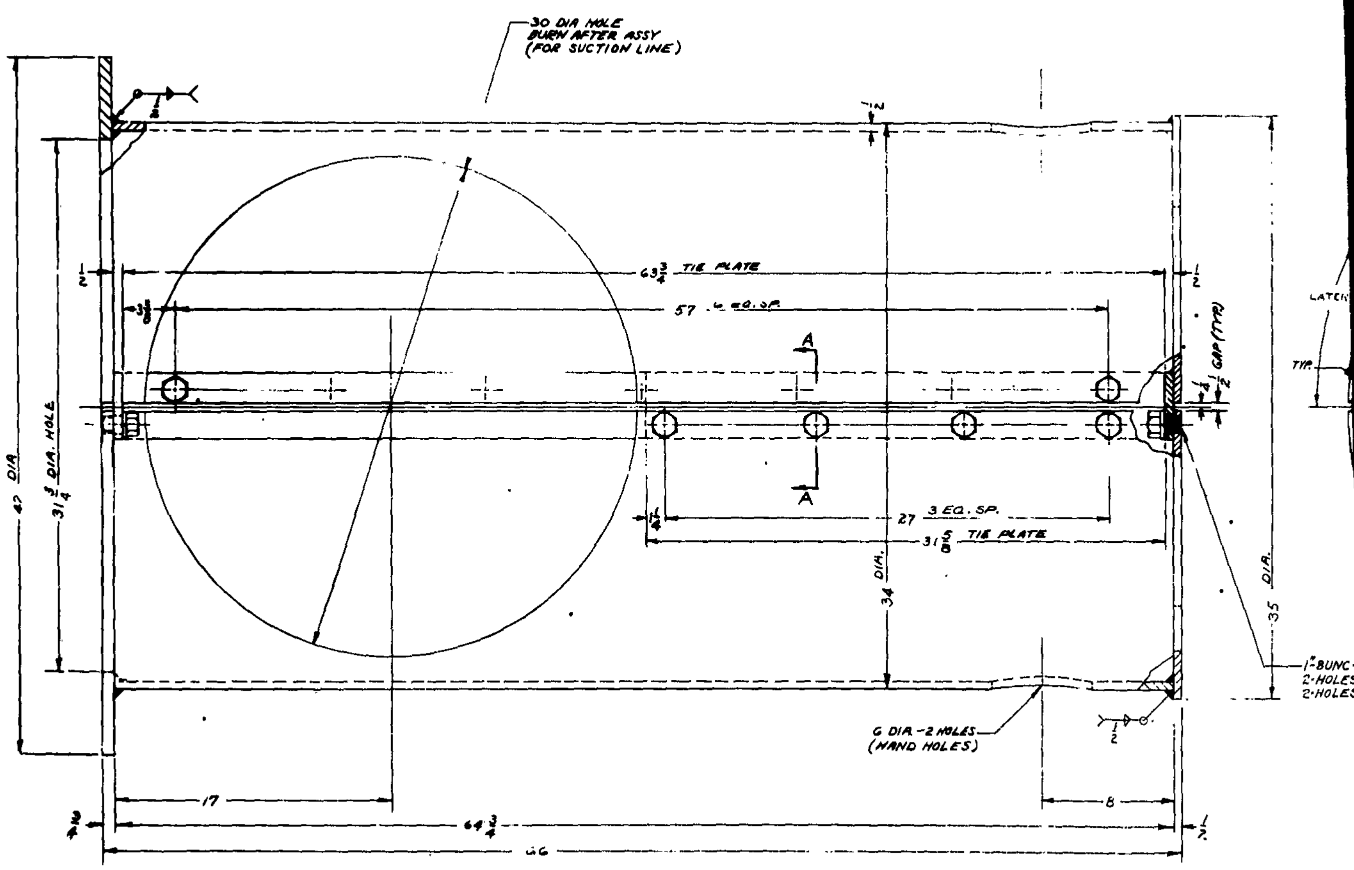
R. LAWRENCE

PRIMARY PUMP & MOTOR OUTLINE

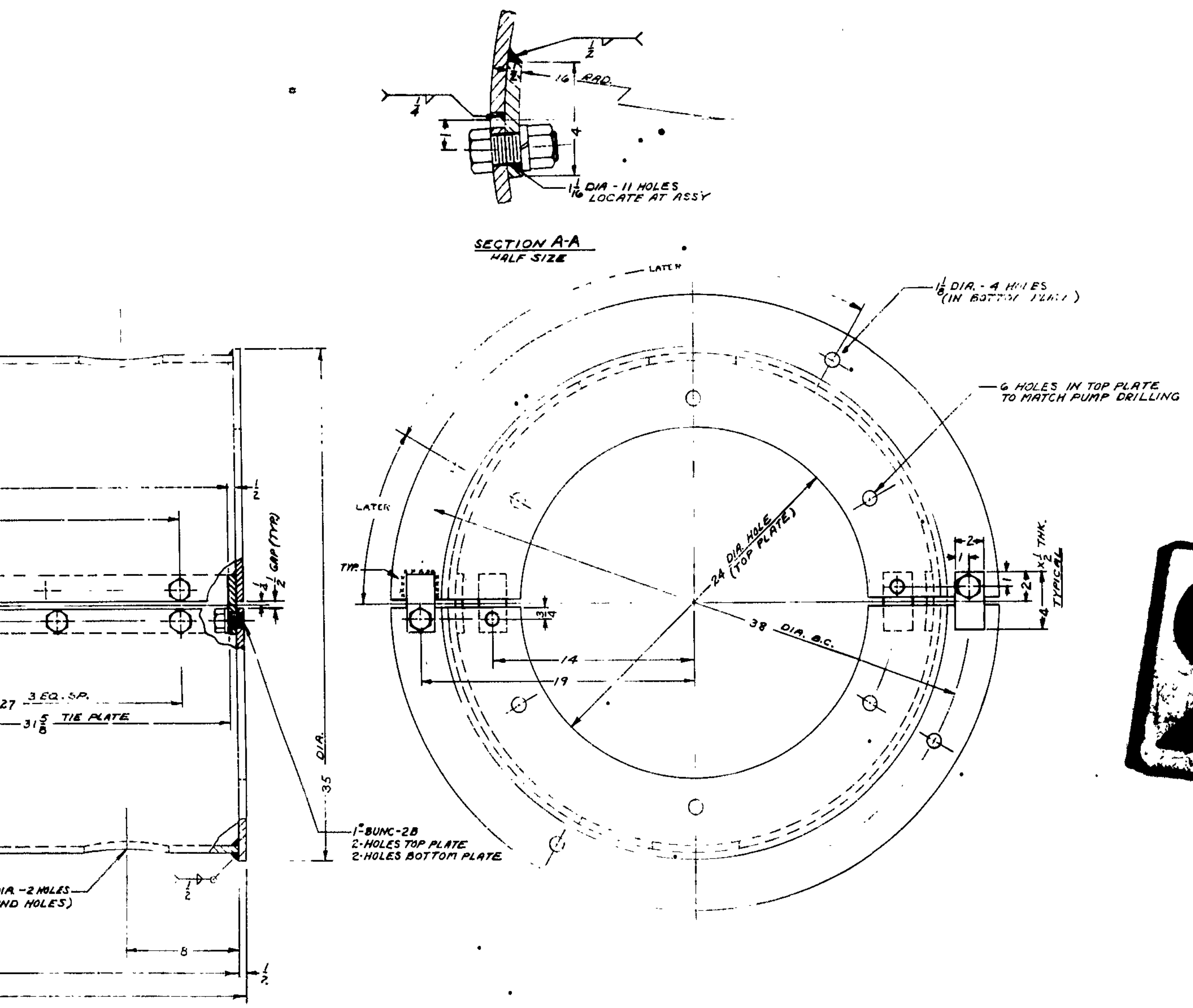
U. S. ARMY
CORPS OF ENGINEERS

M11594-37

1/2" = 1 FT.


30 DIA HOLE
BURN AFTER ASSY
(FOR SUCTION LINE)
63 3/4 TIE PLATE
57 - 4 EQ. SP.
A
A
27 3 EQ. SP.
31 5/8 TIE PLATE
34 DIA.
35 DIA.
42 DIA.
31 3/4 DIA. HOLE
1/2 GAP (TYP)
TYP.
6 DIA - 2 HOLES
(HAND HOLES)
17
8
64 3/4
66

1



M11594-39
A

REF. DWG. M11594-66

MATERIAL
STEEL PLATE ASTM-A242

LIST OF MATERIAL

PRIMARY COOLANT PUMP SUPPORT, UPPER

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-39

9.0 PRIMARY SYSTEM AUXILIARIES

9.1 PRIMARY MAKE-UP SYSTEM

The primary make-up system functions to replace primary system fluid which is blown down, to provide a means for the addition of chemicals to the primary system, and to provide cooling for the control rod drive seals.

Make-up water is added to the primary system through a connection in the 14 in. dia piping between the primary coolant circulating pump and the reactor vessel. Cooling water to the control rod drive seals is added at a constant rate through a manifold to each seal housing. The rate of make-up to the primary system is controlled by the water level in the primary system pressurizer. Make-up and cooling water is mainly demineralized and filtered primary blowdown, with fresh water, also filtered and demineralized, added to replace losses. The make-up water is stored in the primary make-up tank under a blanket of hydrogen at 30 psi. A detailed description of the circuitry and controls is given below, and a line diagram of the primary circuit is shown on Dwg. M-11594-52.

Two constant volume primary make-up pumps, one operating and one on standby, take suction from the primary make-up tank. Discharge is through a filter to two parallel lines, one feeding the control rod drive seals and the other providing make-up to the primary circulating loop.

Each primary make-up pump is valved on both suction and discharge sides to permit isolation for servicing, and each pump discharge line contains a check valve to prevent backflow.

Pressure relief valves are also provided at the discharge of each pump. These valves relieve to the primary make-up tank and function to prevent damage to the pumps in the event that either is started with its discharge valve closed.

Controls are provided to automatically start the standby pump in the event of failure of the operating pump. Annunciation of a pump failure is made at the control console. Indication lights at the control console show which make-up pump is operating. The filter downstream of the make-up pumps functions to collect particles of pump packing or other foreign matter to prevent possible clogging in the control rod drive seal cooling branch.

The branch line to the control rod drive seals contains a pressure control valve. This valve is controlled by differential pressure between primary system and the control rod drive seal branch line and operates to maintain a fixed pressure differential at the control rod drive seals to insure that the direction of flow is into the reactor vessel from the rod drive housing. This line also has a bypass around the differential pressure control valve which contains a standby differential pressure control valve. Globe valves upstream of both pressure control valves permit isolation of either for servicing. The branch line feeding make-up to the primary system relieves to the primary make-up tank through two pressure reducing valves.

Globe, check, and gate valves are located immediately outside of the vapor container on the primary make-up and seal cooling branch lines. The check valves prevent backflow from within the vapor container. The globe and gate valves permit isolation of the check valves.

The need for chemical addition to the primary system is determined by sampling the make-up water in the primary make-up tank, and by analysis of the primary blowdown. A sample cooler is provided for sampling from the primary make-up line downstream of the make-up pumps. A portable pump is used for this service.

Hydrogen is added to the system by use of an addition flask located in a branch line downstream of the primary make-up pumps.

9.2 PRIMARY BLOWDOWN SYSTEM

The primary blowdown system functions to remove impurities suspended in the primary system coolant, and to prevent the buildup of crud and radioactivity which could create hot spots in the primary loop. This is accomplished by continuous removal and treatment of a portion of the primary coolant.

Solids will tend to settle out of the circulating fluid in low velocity areas. The two areas where this condition is most likely to occur are at the bottom of the reactor vessel and the steam generator channel. Consequently, the primary system is blown down at these points. Continuous blowdown is maintained from the bottom of the reactor vessel and the steam generator channel is blown down intermittently. The total blowdown rate is 392 lb/hr.

Fluid taken from the steam generator and the reactor vessel is taken to the outside of the vapor container by separate lines. Globe valves, trip valves, and radiation monitors are connected in these lines immediately outside of the vapor container. The radiation monitors serve as sentinels of the activity of the fluid being blown down, and operate alarms at the control console if the activity reaches dangerous levels. The trip valves are connected so that they will close when a scram is initiated. The globe valves serve to isolate those portions of the lines which contain trip valves and radiation monitors for servicing.

Motor control valves, operated from the control console, regulate the rate of blowdown from each of the lines. Manually operated globe valves in series with the motor control valves serve to isolate these valves for servicing, and may also be used to regulate the blowdown rate.

The blowdown is then passed through the primary blowdown cooler where its temperature is reduced to approximately 110°F. Cooling of the blowdown is required to prevent damage to the demineralizers downstream of the blowdown cooler.

A pressure reducing station is located immediately downstream of the primary blowdown cooler. The pressure of the blowdown is reduced to approximately 125 psi. A bypass line containing a pressure reducing orifice and pressure gauges, plus key-operated globe valves in series with and parallel to the pressure reducing valve,permit isolation of the pressure reducing valve for servicing. A pressure alarm, pH indicator, and local temperature indicator are located immediately downstream of the pressure reducing station. The pressure alarm signals the operator at the control console if high blowdown pressure exists downstream of the pressure reducing station. A pressure relief valve is also provided at this point and vents to the hot waste receiving tank.

A solenoid valve, located immediately downstream of the pressure reducing station, is operated by the conductivity cell downstream of the demineralizer. High conductivity, and the operation of the solenoid are recorded at the control console.

An area type flow meter indicates blowdown rate at this point. A bypass line and isolation valves are also provided for servicing this equipment.

From the flow meter, the blowdown is passed thru a pressure regulating valve to the filter inlet header.

A branch line upstream of the pressure reducing valve diverts some of the flow for oxygen and hydrogen analysis. The diverted fluid is first passed through the oxygen analyzer, and then on through a cooler to the hydrogen analyzer. The fluid is then returned to the filter inlet header.

In addition to the main blowdown line and the diversion line, two other lines feed into the filter inlet header. They are the control rod drive seal leakage line and the fresh water make up line. All lines feeding into the filter inlet header are provided with check valves to prevent backflow.

Filters in parallel take fluid from the filter inlet header and discharge to the demineralizer inlet header. The primary demineralizers are also connected in parallel, and the fluid is passed through them to another header connecting to a second set of filters. These filters pick up any resin carryover from the demineralizers, and deliver filtered and demineralized water to the primary makeup tank.

Each demineralizer and filter is valved on both inlet and outlet to facilitate use of any combination of one upstream and downstream filter and

and one demineralizer. Pressure gauges are provided on the demineralizer inlet and outlet headers, and on the filter outlet header. The demineralizer discharge header is connected to the hot waste drain line through a globe valve. The return line to the primary make-up tank contains a flow meter, pH indicator check valve, and sampling valve in addition to the conductivity cell previously mentioned.

9.3 PRIMARY PURIFICATION MAKE-UP SYSTEM

9.3.1 Primary Purification Loop

The total dissolved and suspended solids in the primary coolant will be maintained below two ppm. This will be accomplished by diverting continuously approximately one gpm (two gpm maximum) of primary coolant from the bottom of the reactor vessel (and intermittently from the bottom of the steam generator) through a purification loop. Only one blowdown line will be operated at a time. In the purification loop the coolant passes successively through a pre-filter, a mixed bed demineralizer, and an after filter. The pre-filter is used to remove most of the suspended solids while the demineralizer removes dissolved as well as any remaining suspended solids. The after filter prevents any resin fines from being carried into the primary system.

9.3.2 Primary System Demineralizer

A nonregenerative demineralizer will be utilized. It is estimated that 2 ft 3 of a nuclear grade mixed bed resin contained in a demineralizer column 1 ft diam and 2.6 ft deep will have enough exchange capacity for at least 1 yr of continuous operation at the normal blowdown rate. With a maximum blowdown rate, the demineralizer will have enough exchange capacity for at least 6 months of continuous operation.

Without a pre-filter, the activity that would accumulate in a demineralizer operating at normal blowdown for one year during the 20th year of plant operation would require 4.4 in. of lead shielding to attenuate sufficiently the radiation from the long-lived radioisotopes. It is expected that the pre-filter will remove most of the long-lived radioisotopes, since they occur chiefly as fine suspended particles in the primary coolant. Therefore, the shielding requirements for the demineralizers should be reduced considerably. Even though the pre-filter may require a considerable thickness of lead shielding, its overall physical dimensions will be much smaller than the demineralizer so that it will be easier to handle. However, final conclusions regarding the effectiveness at the pre-filter will have await the outcome of the SM-1 testing program.

Other demineralizer arrangements were considered, namely (1) a cation column followed by a mixed bed column and (2) a permanently installed mixed bed demineralizer. Since most of the soluble long-lived activity of any concern in the primary water is cationic, a small cation column (which has a much larger exchange capacity for cations per unit volume of resin then does a mixed bed) would be adequate. Therefore, if scheme (1) was used, the mixed bed would serve merely to polish up the water and would require little shielding since it would have relatively little accumulated activity. The cation bed, on the other hand, would require a considerable thickness of lead shielding but since it would be comparatively small, the weight of lead would be considerably less. However, the use of cation and mixed beds would mean that two different resin columns would have to be stocked and disposed of instead of one bed if only a mixed bed demineralizer was used.

In a permanently installed mixed bed demineralizer, fresh and exhausted resin would be sluiced in and out of a permanent resin column. The spent resin would be stored temporarily in an underground tank until it could be drummed up for shipment to a permanent disposal site. While mechanization of the demineralizing system would reduce handling problems considerably, it would also introduce additional piping, valves, tanks, and processing equipment into a plant which is already packed with equipment.

It is felt the objectives of the primary loop purification system can be best served by a combine d mixed bed demineralizer and shipping cask, which would minimize the amount of maintenance and supporting equipment.

9.3.3 Instrumentation for Control of Water Purity

Instrumentation for the control of water purity has been selected on the assumption that a very limited number of personnel would be available for plant operation. Therefore, wherever possible, in-line automatic instrumentation will be installed. This will eliminate many time-consuming laboratory analyses.

9.3.4 Oxygen Analyzer

A continuous sample flow ranging from 50-100 cc/min is diverted from the primary blowdown line (downstream of the pressure reducing station) to an Industrial Instrument oxygen analyzer rated for operation at pressures up to 100 psig. This location gives a more representative primary coolant sample than would a sampling point downstream of the primary demineralizer outlet header where the sample would contain seal leakage return coolant and fresh make-up water as well as primary coolant. The sample stream is returned to the purification loop upstream of the primary demineralizer inlet header so that the metallic thallium ions which have been added by the analyzer will be removed by the demineralizer.

The instrument consists of a mixed bed demineralizer cartridge for cleaning up the sample stream, followed in succession by a conductivity cell, a reaction chamber packed with pure thallium metal, and a second conductivity cell. The dissolved oxygen reacts with thallium to form thallous hydroxide, which increases the conductivity of the water. The conductivity difference is related directly to the oxygen content of the water. A recorder keeps a continuous record of the oxygen concentration and actuates an alarm if the concentration exceeds the prescribed limit.

It has been assumed that about 75 percent of the solids in the primary coolant (which has been limited to 2 ppm total solids) are insoluble. If a sample flow of 100 cc/min contained 0.5 ppm dissolved solids, the resin bed would be exhausted in about 4 months. However, accumulated activity rather than exchange capacity will probably govern the bed's life span. In the event plant operation indicates that activity is accumulated too rapidly in the resin and thus complicating instrument maintenance, a demineralizer will be installed upstream of the analyzer. Connections are provided for one. With a sample flow of 100 cc/min and an oxygen concentration 10 ppb (the maximum allowable in the primary coolant), only 10 percent of the metallic thallium in the reaction chamber will be consumed over a 6 month period.

The reliability and accuracy of Industrial Instrument's oxygen analyzer will be verified at SM-1 in the near future. If the testing program there indicates the unsuitability of this analyzer, another instrument can be substituted at the sampling point without introducing any significant changes in plant design. The selection of Industrial's analyzer seems justified in view of its simplicity and the possibility of reducing the volume of liquid wastes considerably by returning the sample stream to the primary system without introducing any supporting equipment (e.g., a pump, a collection tank and a hydrogen purging system for the latter).

9.3.5 Hydrogen Analyzer

There are no commercial analyzers available which continuously monitor dissolved hydrogen in the concentrations (15-30 cc/1) that will be maintained in the primary coolant. In the absence of a commercial analyzer, a method of hydrogen analysis developed at KAPL will be adapted for use in the SM-2. While the KAPL method is not automatic, the analysis is considerably less time-consuming than conventional laboratory methods, and its accuracy compares favorably with the latter. The analytical method may be described briefly as follows:

(1) Sample bomb of known volume is filled with a pressurized sample of hydrogenated water which has been cooled to ambient temperature. The sample will be drawn from the oxygen analyzer sample line.

(2) The water in the sample bomb is allowed to expand by cracking a valve. The expansion arises from the decompression of the water sample and the release of gasses dissolved in excess of saturation at atmospheric pressure. The gases collect as a gas bubble in the top of the sample bomb.

The volume of dissolved gases coming out of solution may be expressed by:

$$V_b = E - C_W$$

where

V_b = volume of dissolved gas coming out of solution, cc

E = expansion volume, cc

C_W = increase of same volume, due to decompression, cc

For operation of this instrument the expansion due to decompression of the water will be negligible because the sample is drawn downstream of the pressure reducing station in the primary blowdown line where the pressure will probably not exceed 100 pis.

(3) The expansion is measured by allowing the water sample to expand into a measuring burette.

(4) A known volume of air is circulated through the sample bomb.

(5) The fraction of hydrogen in the known volume of air is determined by pumping it through a Cambridge hydrogen analyzer, which operates on the thermal conductivity principle.

Knowing the sample volume, the volume of gas in the sample bomb (the gas collected in the top of the bomb), the volume of air being circulated through the sample, and the fraction of hydrogen in the air, the dissolved H_2 concentration can be calculated by

$$H_2,\ cc/1 = (V_a + V_b)\ F \times \frac{1000}{V_s - E}$$

where

V_a = volume of air circulated through sample, cc

V_b = volume of gas (bubble at top of bomb) in bomb, cc

F = fraction of hydrogen in air as determined by the Cambridge analyzer

V_s = volume of sample, cc

E = expansion volume, cc

To the concentration of H_2 calculated by the above equation must be added the residual H_2 in the bomb sample at atmospheric pressure and at the prevailing temperature.

9.3.6 Conductivity Metering

Industrial Instruments conductivity cells capable of operating at temperatures up to 212°F and pressures to 100 psig are installed downstream of the after filter outlet header to detect the break-through point of the primary demineralizer and in the primary blowdown line to monitor the conductivity of the primary coolant.

The conductivity cell is connected to an indicator controller. A clock-type dial indicates the conductivity at the sampling point. An automatic cell selector switch checks the sampling points sequentially, about 1 minute being required for each point.

When the conductivity of a sample stream exceeds the control setting, it is indicated by a signal light on the automatic cell selector panel board as well as by an audible alarm. Conductivity sampling points in the secondary system will be tied in to the same selector switch.

9.3.7 pH Metering

Beckman pH cells will be installed on the discharge side of the pressure reducing valve and on the outlet side of the after filter header in the primary blowdown line.

The pH cells in the primary and secondary system will be tied in to one continuous indicating meter through a sequential automatic selector switch. About 20 sec will be required for each sample point. A combined audible alarm and signal light system will alert the plant operator when the pH falls outside the prescribed range.

9.3.8 Chemical Additives

9.3.8.1 Hydrazine Addition

At startup, 2 gal of 2 percent (by volume) hydrazine solution is added to the primary system to scavenge oxygen present in the demineralized water used to fill the system. Following startup, oxygen can enter the primary system from leakage of air through pumps and valves, primary in the reactor. Oxygen from the first two sources should be small and hydrogen gas present in the primary coolant (15-30 std. cc/l) combines immediately with any oxygen present under the catalytic action of gamma radiation in the core. Therefore, very little additional hydrazine will be required.

If required, further batches of 2 percent (by volume) hydrazine solution will be added until the oxygen analyzer shows that the oxygen concentration is being maintained below 0.01 ppm by the combined action of the dissolved hydrogen and the gamma flux.

The hydrazine is introduced into the primary system from a 5 gal tank connected to the inlet manifold of the primary make-up pumps.

9.3.8.2 Boric Acid Addition

Boric acid will be injected into the primary system only when malfunctioning of the control rod system makes it impossible to shut down the reactor plant.

When boric acid addition becomes necessary, the solution is prepared in a 200 gal stainless steel tank equipped with a heating element and a mechanical stirrer. About 520 lb of boric acid (twice the amount needed in the primary loop to keep the reactor subcritical at 68 F) will be dissolved in 200 gal of water heated to 203 F.

The boric acid is added at the suction side of the primary make-up pump. Prior to addition, the blowdown will be increased to 2 gpm if it is not already operating at this rate. The appropriate valves will then be turned to cut off flow from the primary make-up tank and start boric acid addition from the 200 gallon feed tank. As boric acid injection starts, the blowdown purification section will be valved off and the blowdown will be diverted to the hot waste system by opening the valve to the hot waste dump line. When sufficient boric acid has been injected, the feed tank line will be turned off, and flow from the primary make-up tank will be resumed. At the same time, the hot waste dump line will be closed and the blowdown will be diverted around the demineralizers through the by-pass line.

The primary make-up pump will inject boric acid into the primary loop at the rate of 2 gpm. Therefore, 100 gal of boric acid solution containing 260 lb of boric acid (the amount required for shutdown at 68 F) can be injected in about 50 minutes. Auxiliary power is available for the make-up pump if the regular power source fails.

9.3.9 Auxiliary Clean-up Systems

Provision has been made to reduce the raioactivity and turbidity of the upper shield tank water during a core change. The spent fuel tank will also be provided with a clean-up system.

The upper shield tank clean-up system may be briefly described as follows:

Four drainage holes (90° between centers) are located in the bottom of the upper shield tank around the reactor vessel flange. Piping connects these drainage holes to a circular manifold around the neck of the reactor vessel in the lower shield tank. The manifold is connected to the suction side of a pump. The water will be pumped to the top of the tank through a filter and a demineralizer.

Reduction of the amount of radioactive crud entering the upper shield tank and the resulting turbidity will be accomplished by operating the upper shield tank clean-up system and primary blowdown line simultaneously during a core change. A booster pump will be tied into the primary blowdown line downstream of the pressure reducing valve. It will pump water from the bottom of the reactor vessel at the rate of 10 gpm through both demineralizer lines to the primary make-up tank. Both primary make-up pumps adjusted to operate at maximum capacity will return 10 gpm to the primary loop.

Normally, the spent fuel tank water will be circulated through a filter to remove crud from it. If however, the water becomes extremely radioactive, a demineralizer may be connected to the clean-up system.

9.3.10 Decontamination

The build up of activity on system surfaces and components can seriously restrict accessibility and prevent necessary maintenance. Therefore, rapid and efficient decontamination is required to reduce the activity in the plant to safe working levels. The SM-2 decontamination procedure given here is based on the conclusions of APAE Memo No. 234[1].

The procedure involves the following liquids applied as described below. The volume of each cycle is approximately 1520 gallons.

1. A caustic permanganate solution (10 percent $NaOH$ plus 5 percent $KMnO_4$)

2. A water rinse, using condenser water

3. A citrate combination solution (5 percent ammonium citrate, 2 percent citric acid, 0.5 percent versene)

4. Three water rinses, the first using condenser water and the last two demineralized water from the primary make-up tank.

Decontamination of the primary system is accomplished utilizing the following procedure:

1. The primary system is cooled and depressurized.

2. The reactor vessel cover is removed and stored in the upper shield tank. Fuel elements and control rods are removed and stored in the spent fuel storage tank.

3. A temporary cover is secured to the reactor vessel. This cover is penetrated by piping for filling and venting the primary system.

4. Drainage taps located in the lowest points in the steam generator and reactor vessel blowdown lines, under the pass plate of the steam generator, and the two low points of the primary piping are connected by means of temporary piping to the decontamiation pump. Refer to Dwg. F9-54-1040 for the location of these taps and details.

5. The primary coolant water is pumped into the waste disposal system through the sump pump line, as shown in Dwg.

6. The valves to the primary system are closed and the air in the system is partially evacuated through the vent line in the temporary reactor vessel head by means of a vacuum pump.

7. The primary system is filled with alkaline permanganate (the first decontamination solution) which is pumped by a portable pump from a 1500 gal mixing tank into the reactor vessel through the fill line.

8. Back flushing of the primary coolant canned rotor pump bearings and the control rod drive seals with demineralized water is started, since they cannot be exposed to the decontamination solutions. The seal leakage line to the primary purification loop is valved off, but temporary piping allows water from this line to drain into the sump pit. Temporary piping is connected to the primary make-up line to divert demineralized water to the primary coolant pump bearings and control rod drive seals.

9. After the system has been filled, the primary pump is turned on and off until the air remaining in the steam generator is flushed out into the reactor where it is vented through the vent line in the temporary reactor vessel cover. Filling is continued until the primary system is completely filled.

10. The valve in the temporary piping from the reactor blowdown line to the waste tank is adjusted so that the flow through it equals the flow rate of demineralized water through the control rod drive seals and the primary pump bearings.

11. Circulation of the decontamination solution is then started, using the primary coolant pump. Steam is passed through the secondary side of the steam generator in order to heat the decontamination solution.

12. After the decontaminating solution has been allowed to cool, it is pumped into one of the 5000-gal waste storage tanks.

13. Subsequent flushes of the primary system (citrate solution and rinses) are accomplished as described in steps six through twelve above.

Compliance with this regulation will be accomplished by diluting the gases released from the storage cylinders with a stream of air from a 10000 cfm fan.

9.3.11 Sampling Points

In order to verify the analysis of the in-line chemical instrumentation, as well as to obtain additional information, samples of primary coolant will be withdrawn periodically for laboratory analysis. These and other sample points from associated systems are listed below.

1. Intake side of hydrogen analyzer (with sample cooler).

2. Downstreams of after filter manifold in purification loop (no sample cooler).

3. Primary make-up line just downstream of make-up tank (with sample cooler).

4. Lower shield tank drainage line (no sample cooler).

5. Discharge side of pumps in upper shield tank and spent fuel tank clean-up systems (no. sample coolers).

6. Discharge side of vapor container cooling water pumps (no sample cooler).

9.3.12 Corrosion Inhibitor for Lower Shield Tank Water

Potassium chromate (K_2CrO_4) is added to the lower shield tank water in order to inhibit corrosion. Enough K_2Cr_4 is added (52 lb to give a concentration of 1000 ppm chromate ion (CrO_4) in the lower shield tank water.

Batches of chromate solution are poured down the air vent pipe that extends from the top of the upper shield tank to the lower shield tank until the required amount (52 lb of K_2CrO_4) has been added. The pH is adjusted to about 8.5 by adding potassium hydroxide. The tank is filled by opening the gate valve on the branch of the fresh water make-up line that terminates at the top of the air vent pipe. When the tank is filled, samples are withdrawn and checked for pH and chromate ion concentration.

1000 ppm of chromate will prevent practically any corrosion of the steel. This is probably considerably higher than the amount actually needed. However, because severe local pitting will occur if the chromate concentration falls below a certain minimum (which is a function of many conditions), a high concentration is recommended. Potassium chromate is used rather than the lithium or sodium forms because from a radiological point of view, it is the least hazardous of the three.

9.3.13 Chemical Laboratories

Space has been provided in the primary building for adjoining hot and cold laboratories. Each laboratory will have floor and wall cabinets for the storage of analytical equipment chemicals, glassware, books, and records. Hoods, sinks and benches are also provided for the work being conducted in the laboratories. Primary water samples will be taken directly from the purification loop instead of running sample lines to the hot laboratory. This is being done to limit as much as possible the hot areas in the primary building. However, all secondary sample lines will come into the cold laboratory to facilitate laboratory work.

9.4 SECONDARY BLOWDOWN SYSTEM

The secondary blowdown system functions to prevent the accumulation of solids in the secondary (feedwater) side of the steam generator, and to minimize fouling of the heat transfer surfaces. This is accomplished by continuous removal or blowdown of a portion of the fluid in the steam generator shell. The blow down, during normal plant operation, is sent through temperature and pressure reducing processes and dumped in the storm sewer (plant waste drain). If the blowdown becomes contaminated, it is diverted to the hot waste receiving tank. A detailed description of the processing of the blowdown is given below.

Secondary blowdown is taken from the steam generator continuously at a point just below the operating water level in the shell side of the vessel, and intermittently from the bottom of the shell compartment. The fluid from each extraction point is taken to the outside of the vapor container through separate lines.

Immediately outside of the vapor container, globe and solenoid operated trip valves are provided on each line. The trip valves are actuated when a scram is initiated to prevent flow out of the vapor container. The globe valves serve to isolate the trip valves for servicing.

Motor operated valves, downstream of the trip valves regulate the rate of blowdown from each steam generator extraction point. These valves are operated from the control console.

Manually operated globe valves are located downstream of the motor operated valves. These valves serve to isolate the motor operated valves for servicing, and may also be used to manually regualte the blowdown rate.

The blowdown from each extraction point is combined downstream of the motor operated valves and is then passed through the secondary blowdown cooler. This is a heat exchanger which functions to lower the temperature of the fluid to below the boiling point. A bypass line is provided around the blowdown cooler, and three valves, two in line with the cooler and one in the bypass line, permit isolation of the cooler for servicing.

Immediately downstream of the secondary blowdown cooler, the blowdown passes through a pressure reducing station. The pressure is reduced to approximately 50 psi downstream of this station. A spring loaded pressure reducing valve is used to accomplish the pressure reduction. A bypass line, similar to that for the blowdown cooler, and valves are provided for isolation of the pressure reducing valve for servicing. A globe valve in the bypass line with pressure gauges upstream and downstream, permits manual regulation of pressure reduction when the pressure reducing valve is isolated.

A pressure relief valve is located downstream of the pressure reducing station. Excessive pressures downstream of the station will cause the relief valve to open and discharge to the hot waste receiving tank.

A solenoid operated valve, for bypass of the secondary blowdown, if contaminated, is also provided at this point. High activity of the blowdown fluid will initiate an alarm at the control console, after which the operator actuates the solenoid operated valve. Contaminated fluid is diverted to the hot waste receiving tank.

Downstream of the pressure reducing station the blowdown passes through a conductivity cell and flow meter and discharges to the storm sewer. The conductivity cell records the level of impurities in the blowdown at the control console. Blowdown rate may be varied in order to maintain a high level of system purity. The flow meter indicates the blowdown rate at the control console. This equipment is also bypassed and valved to permit isolation for servicing.

Local and remote temperature indication are located respectively upstream and downstream of the pressure reducing station. The remote temperature indicator monitors performance of the secondary blowdown cooler. Temperature downstream of the cooler is indicated at the control console.

9.5 VAPOR CONTAINER COOLING WATER SYSTEM

The vapor container cooling water system provides cooling for the upper and lower shield tanks, the primary (coolant) circulating pump, and the space cooler within the vapor container, plus the primary and secondary blowdown cooler outside of the vapor container.

The cooling water is stored in the vapor container cooling water tank. This tank is a horizontally mounted unit with 625 gal capacity, and is vented to the atmosphere. Two vapor container cooling water pumps, one operating and one on standby, take suction through a strainer from the cooling water storage tank, and discharge through check and gate valves to the cooling water supply line. Five parallel circuits carry the cooling water through the coolers listed above and deliver it to a common return line. The circulating fluid is then passed through the cooling water cooler. This is a heat exchanger which removes the heat that the fluid has received in passing through the five coolers. The circulating fluid is then returned to the vapor container cooling water storage tank.

Feed lines to cooling equipment within the vapor container are equipped with globe valves and check valves outside of the vapor container wall. The check valves prohibit back flow or flow out of the vapor container.

Solenoid operated trip valves are located outside of the vapor container on return lines from equipment within the container. The valves are automatically closed by a scram to prevent flow out of the vapor container. Globe valves are provided on the return lines from each cooling component for balancing and apportioning flows in accordance with the component requirements. Relief valves on the return lines from each cooling component are provided. Gate valves on the inlet of the shield tank cooling coils, and the space cooler are provided to isolate either of these branches for servicing. Key operated valves are provided on both the inlet and outlet cooling water lines for the primary coolant circulating pump.

The return line from the primary coolant circulating pump cooling coil contains remote temperature and low flow indicators. High temperature or low flow will initiate an alarm at the control console indicating that inadequate cooling is being provided for the pump.

Each cooling water circulating pump develops a TDH of 127 ft while delivering 120 gpm of cooling water. Only one pump operates at a time, and controls are provided to start the standby pump if the operating pump fails. Indication of which pump is in operation is shown on the control console.

The heat exchanger duties for the components of the system are as follows:

Component	Duty (Btu/hr)
Upper shield tank coil Lower shield tank coil	225,000
Primary circulating pump coil	450,000
Space cooler	30,000
Primary blowdown cooler	440,000
Secondary blowdown cooler	267,000
Cooling water cooler	1,412,000

Samples of the circulating fluid will be taken periodically to determine activity, and fluid may be diverted to the hot waste tank if contaminated.

Carbon steel piping is used throughout the system. Piping at the discharge of the cooling water circulating pumps is 2 1/2 in. diam and is stepped down in size as branch lines are tapped off of the main line.

The design pressure for the system is 150 psi.

9.6 SPENT FUEL TANK RECIRCULATION SYSTEMS

The spent fuel tank recirculation system function to maintain purity of the water in the spent fuel tank, and to provide either heating or cooling for this water as required by the site and/or operating conditions. These tasks are accomplished by circulation of the water through a filter and either an electric heating unit or an exchanger for cooling. The water is then returned to the spent fuel tank.

A centrifugal pump is used to circulate the water. The pump takes suction from the bottom of the spent fuel tank through a foot valve and a strainer. Gate valves are provided on both the suction and discharge sides of the pump to permit isolation for servicing. A check valve is provided on the discharge side of the pump to prevent backflow.

The electric heating unit and the cooling unit are connected in parallel and valved so that flow may be directed through the proper unit. A local temperature indicator upstream of these units indicates which process is required.

Connections for a demineralizer are provided downstream of the heating and cooling units. A sample point is located downstream of the filter. If the filter fails to produce the required degree of purity of the water, the demineralizer can be connected to assist in the treatment of the circulated water. A second sample point downstream of the filter permits analysis of the water after it has passed through the demineralizer. If the required purity is still not realized, a diversion valve is provided for dumping the water to the hot waste tank.

Make-up water for the spent fuel tank is supplied form the primary make-up tank.

9.7 SHIELD TANK RECIRCULATION SYSTEM

The shield tank recirculation system is provided to maintain purity of the water in the upper shield tank. The system consists of a centrifugal pump, filter, demineralizer, interconnecting piping, and valves.

The pump takes suction from the bottom of the upper shield tank and passes the water through the filter and demineralizer and returns it to the shield tank.

If the shield tank water becomes highly contaminated, the flow from the pump discharge may be diverted to the hot waste tank. A valve, downstream of the demineralizer, is provided for sampling to determine purity.

Valves upstream and downstream of the pump permit isolation for servicing. The downstream valves also permit diversion of flow to the hot waste tank. A check valve on the discharge side of the pump prevents backflow in the system.

Valves are also provided on each side of the demineralizer to isolate it for replacement.

9.8 VAPOR CONTAINER VENTILATION SYSTEM

The vapor container ventilation system functions to remove radioactive particles from the air within the vapor container, and to ventilate the vapor container prior to entry by personnel. Reactor shutdown is a prerequisite to the operation of this system.

The system consists of an air filter and a blower together with its associated piping. The blower takes suction from the vapor container through the air filter and discharges the filtered air back into the vapor container. This cycle is repeated until the activity of the air within the vapor container has been reduced to a safe level.

After the air within the vapor container has reached acceptable radiation levels, the blower discharge is diverted to the vent stack. At the same time the vapor container access door is opened,allowing fresh air to be drawn into the vessel.

Gates valves are located on both the suction and discharge lines, and are closed during normal plant operation. These valves are only opened after the plant is shut down, and the pressure within the vapor container is normal.

Valves on the filter, blowe, and vent stack are provided to drain collected condensate to the hot waste tank.

9.9 HOT WASTE DISPOSAL SYSTEM

The hot waste disposal system functions to collect, store, and process wastes which are known to be or sustpected of being radioactive, including fluids collected as a result of decontamination activities.

Wastes external to the vapor container are drained to and collected in the 2500 gal capacity hot waste tank. Wastes from within the vapor container are first collected in a sump and then pumped to the 2500 gal tank. Globe valves with a solenoid operated trip valve between them are located in this line immediately outside of the vapor container. The trip valve closes when a scram is initiated to prevent flow out of the vapor container. The globe valves serve to isolate the trip valve for servicing. From the 2500 gal tank, the wastes are pumped to either of two 5000 gal tanks for temporary storage.

Further processing of the wastes involves reduction of volume and concentration of contaminants. An evaporator condenser is used for this purpose. Wastes are pumped from the 5000 gal tanks to a 200 gal holding tank and then to the evaporator condenser. Concentrated contaminants are removed from the condenser and stored in shielded drums for shipment. Condensate from the reduction process is passed to a 300 gal condensate tank and is then mixed with secondary system condensate as it is discharged to the plant drain.

Radioactive gasses which build up in the hot waste system are pumped through an absolute filter into storage cylinders by a compressor. After activity levels of these gasses have reduced to safe levels, they are discharged through a blower to the vent stack.

The waste in the 2500 gal tank may be sampled prior to discharge to the 5000 gal tanks. If acticity is low, the flow may be diverted through a demineralizer to the plant drain.

The condensate line from the evaporator condenser contains valves which will permit further activity reduction by bypassing the condensate through a demineralizer.

Steam is used for the evaporator, and for heating of the wastes stored in the 5000 gal tanks.

Globe valves are used in all lines to permit operator control of the treatment process, and check valves are provided in all lines where reversal of flow is prohibitive.

9.10 EQUIPMENT LIST

EQUIPMENT LIST
AUXILIARY EQUIPMENT
(PRIMARY)

Reactor, Steam Generator & Auxiliary Heat Exchanger

Item	Service	Quantity	Flow	TUBE SIDE Oper. &Design Press.	Oper. Temp. In/Out ^oF	Flow	SHELL SIDE Oper. & Design (PSIA)	Oper. Temp. In/Out ^oF	Remarks
	Primary Blowdown Cooler	1	2.6 gpm	2000/2200	515/110	29.4 gpm	75/112	100/30	Rated @ 4.4 x 10^5 BTU/hr Design is tube in tube helicoil of 304 SS w/SS manifold.
	Secondary Blowdown Cooler	1	600 #/hr	480/950	452.8/110	17.8 gpm	56/84	100/130	Rated @ 2.6 x 10^3 BTU/hr Design is tube in tube helicoil of copper w/bronze manifold
	V.C. Cooling H_20 Cooler	1	108 gpm	40/75	130/100	108 gpm	40/75	70/100	Rated @ 1.61 x 10^6 BTU/hr Counter flow design w/copper tube in C. S. shell
	Spent Fuel Pit Cooler	1	10 gpm	35/75	150/120	10 gpm	35/75	70/100	Rated at 1.3 x 10^3 BTU/hr Tube in tube parallel thru design
	Primary Auxiliary Skid	1	Width 9' 4-3/8"	Length 19'	Weight 19,699				

EQUIPMENT LIST
AUXILIARY EQUIPMENT
(PRIMARY)

PUMPS

SERVICE	QUANTITY	TYPE	FLOW-GPM	DIFFERENTIAL HEAD	TEMPERATURE	H. P.
Primary Make-Up Pump	2	Positive Displacement	0 - 6.2	2470 psi	90^{0} F	10
V. C. Cooling H_20 Pump	2	Horiz. Centrifugal	120	55 psi	100^{0} F	7.5
Spent Fuel Fit Pump	1	Horiz. Centrifugal	15	43 psi	150^{0} F	2
Shield Tank Circ. Pump	1	Horiz. Centrifugal	25	27 psi	150^{0} F	2
V. C. Sump Pump	1	Vert. Centrifugal	10	43 psi	140^{0} F	1.5
Waste Disposal 2500 Gal. Tank Pump	1	Vert. Centrifugal	25	75 ft.	150^{0} F	2.0
Waste Disposal 5000 Gal. Tank Pump	2	Vert. Centrifugal	10	75 ft.	150^{0} F	2.0
Waste Disposal Condensate Pump	1	Horiz. Centrifugal	5	75 ft.	100^{0} F	1.0

TANKS

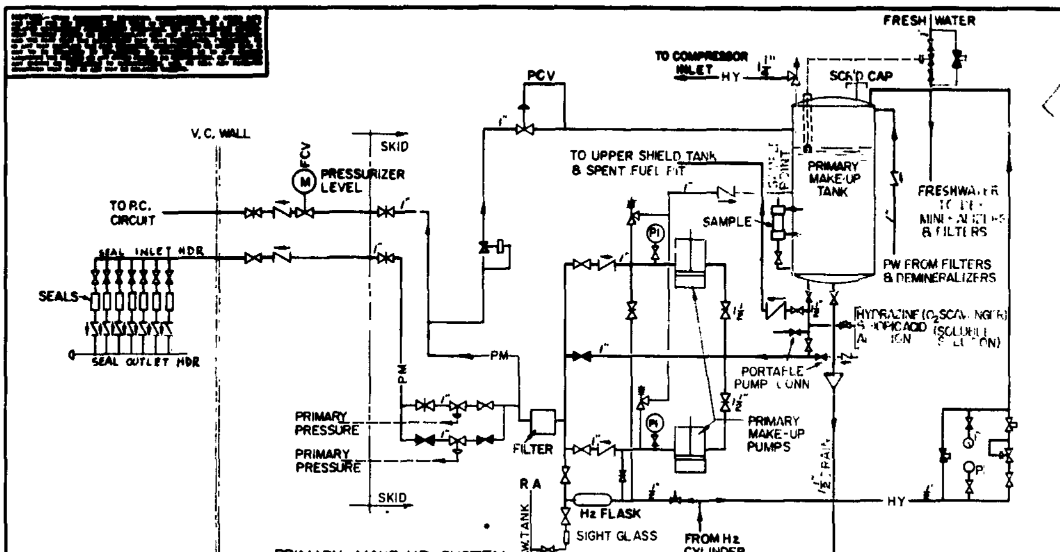
SERVICE	QUANTITY	CAPACITY	DIMENSIONS	DESIGN PRESSURE	MATERIAL
Primary Make-up Tank	1	500 gal.	4' Dia. x 5' long	30 psig	304 S.S.
V.C. Cooling H_20 Storage Tank	1	625 gal.	3' 6" Dia. x 8' long	Atmos.	Carbon Steel
Hot Waste Tank	2	5000 gal.	7' Dia. x 17' long	Atmos.	Carbon/w ceramic liner
Hot Waste Tank	1	2500 gal.	6' Dia. x 14' long	Atmos.	Carbon/w ceramic liner
Spent Fuel Tank	1	4700 gal.	9' Dia. x 20' high	Atmos.	C. S. w/18 Gr. Cadmium Liner
Holding Tank	1	200 gal.	2-1/2' Dia. x 6' high	Atmos.	304 S.S.
Condensate Tank	1	300 gal.	3' Dia. x 6' long	Atmos.	Carbon Steel

EQUIPMENT LIST
AUXILIARY EQUIPMENT
(PRIMARY)

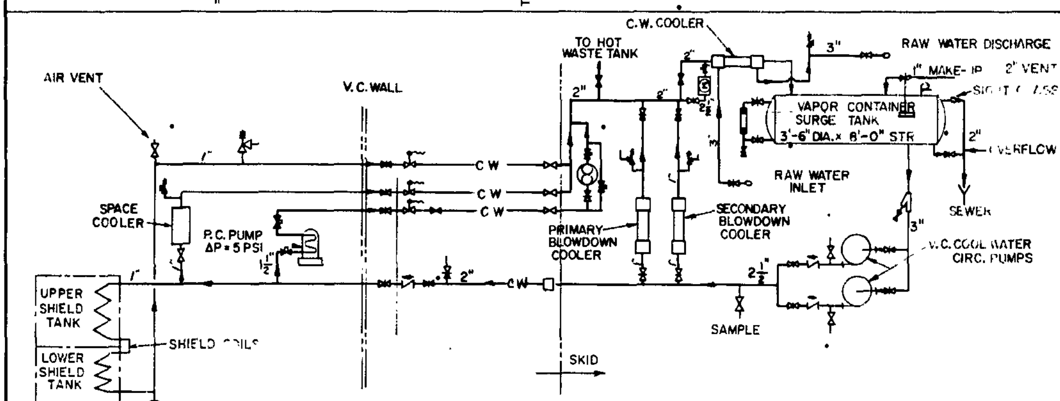
SERVICE	QUANTITY	PRESSURE OPER./DESIGN	OPER. TEMP ^{0}F			
				Flow	H. P.	
Ventilation Fan	1	8" H_20	100	2300 CFM (@ 14.7 psia & $60^{0}F$)	5	
Ventilation Fan (Hot Waste System)	1	1" H_20	100	10000 CFM (@ 14.7 psia & $60^{0}F$)	5	
		Pore Size	Pressure Drop	Temp.	Material	
Ventilation Filter (Hot Waste & Ventilating System)	2	0-3 microns	4" H_20	$100^{0}F$	Glass	
		Flow	Pressure	Temp.		
Demineralizer	3	10 gpm/Ft^2	50 psig	120^{0} F		
				Stages		
Air Compressor (Hot Waste System)	1	10 CFM	1000 psia	3		
				Steam Size	Cooling Water	
Evaporator (Waste Disposal)	1	50 gph	50 psia	Flow 500 #/hr @ 100 psig	720 gph 70^{0} - $155^{0}F$	10" H_20 Vacuum in Condenser

REFERENCES

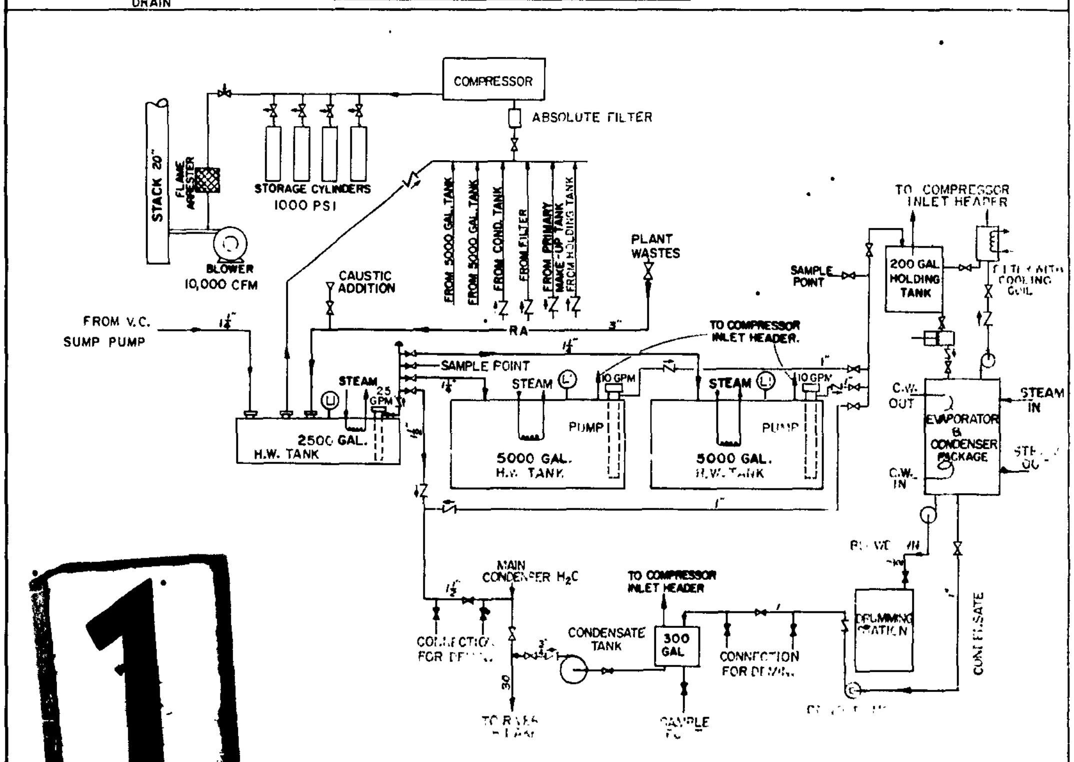
1. Pancer, G. P., "Engineering Study of Methods for Decontamination of SM-2, "APAE Memo No. 234, February 15, 1960.



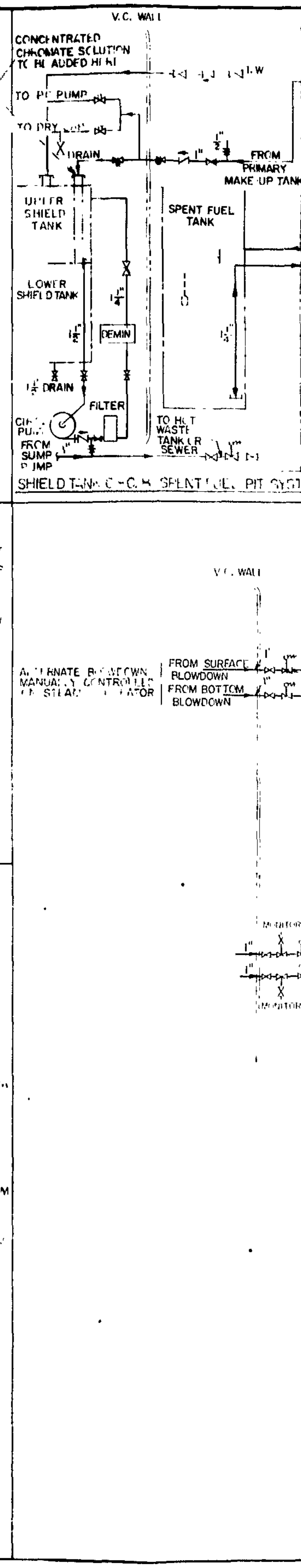
PRIMARY MAKE-UP SYSTEM

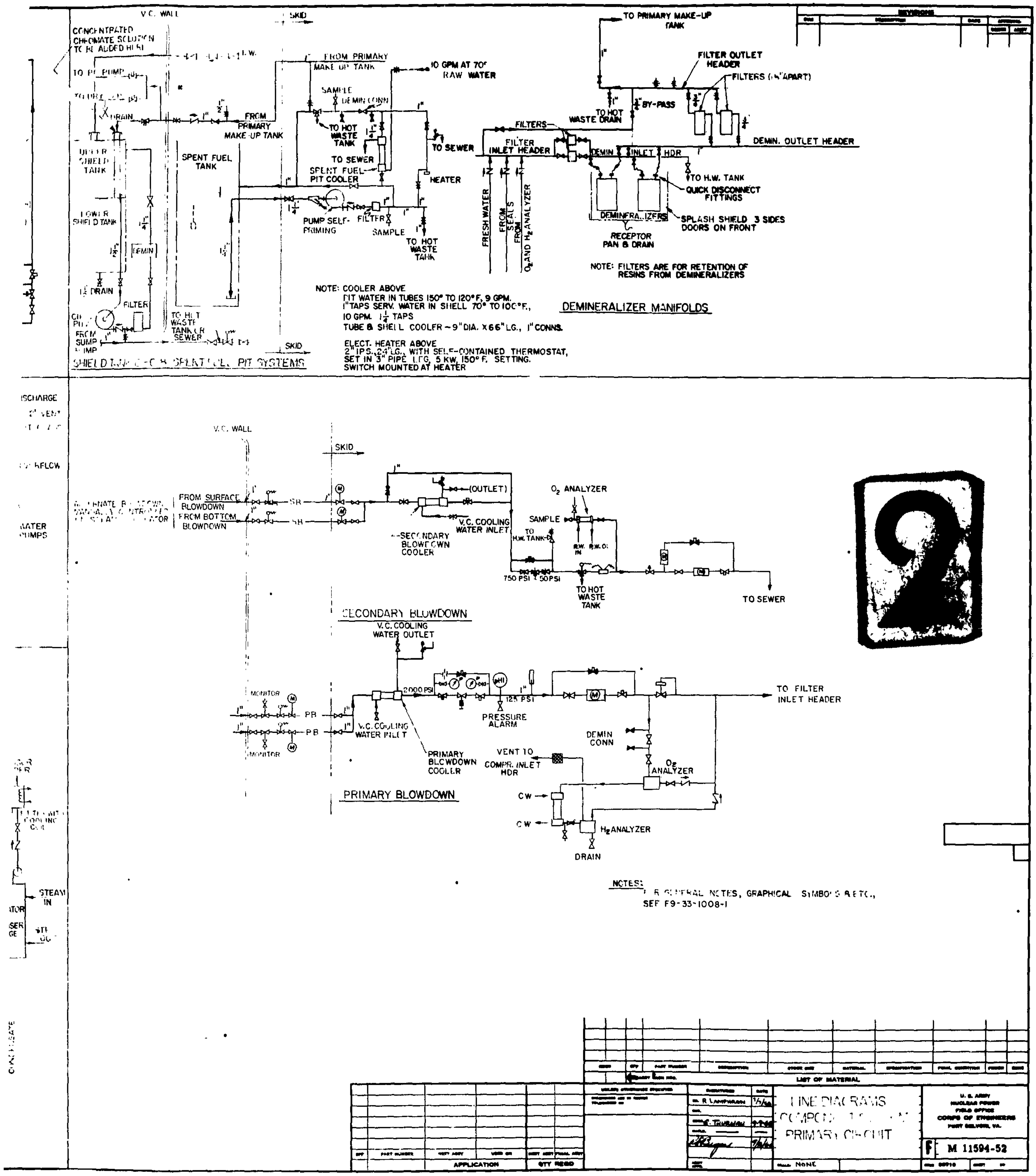


VAPOR CONTAINER COOLING WATER SYSTEM



WASTE DISPOSAL SYSTEM




V.C. WALL
SKID
CONCENTRATED CHROMATE SOLUTION TO BE ADDED HERE
TO PRI PUMP
FROM PRIMARY MAKE UP TANK
10 GPM AT 70° RAW WATER
SAMPLE
DEMIN CONN
DRAIN
FROM PRIMARY MAKE-UP TANK
TO HOT WASTE TANK
TO SEWER
UPPER SHIELD TANK
SPENT FUEL TANK
SPENT FUEL PIT COOLER
HEATER
LOWER SHIELD TANK
PUMP SELF-PRIMING
FILTER
SAMPLE
TO HOT WASTE TANK
DEMIN
DRAIN
FILTER
TO HOT WASTE TANK OR SEWER
FROM SUMP PUMP
SKID
SHIELD TANK & SPENT FUEL PIT SYSTEMS
NOTE: COOLER ABOVE
PIT WATER IN TUBES 150° TO 120°F, 9 GPM.
1" TAPS SERV. WATER IN SHELL 70° TO 100°F,
10 GPM. 1¼" TAPS
TUBE & SHELL COOLER ~ 9" DIA. X 66" LG., 1" CONNS.
ELECT. HEATER ABOVE
2" IPS., 24"LG., WITH SELF-CONTAINED THERMOSTAT,
SET IN 3" PIPE LEG, 5 KW, 150°F. SETTING.
SWITCH MOUNTED AT HEATER
TO PRIMARY MAKE-UP TANK
FILTER OUTLET HEADER
FILTERS (18" APART)
TO HOT WASTE DRAIN
BY-PASS
FILTERS
FILTER INLET HEADER
DEMIN. OUTLET HEADER
DEMIN INLET HDR
TO H.W. TANK
QUICK DISCONNECT FITTINGS
FRESH WATER
FROM SEALS
FROM O2 AND H2 ANALYZER
DEMINERALIZERS
SPLASH SHIELD 3 SIDES DOORS ON FRONT
RECEPTOR PAN & DRAIN
NOTE: FILTERS ARE FOR RETENTION OF RESINS FROM DEMINERALIZERS
DEMINERALIZER MANIFOLDS
V.C. WALL
SKID
FROM SURFACE BLOWDOWN
FROM BOTTOM BLOWDOWN
SB
(OUTLET)
O2 ANALYZER
SAMPLE
V.C. COOLING WATER INLET
SECONDARY BLOWDOWN COOLER
TO H.W. TANK
R.W. IN
R.W. OUT
750 PSI
50 PSI
TO HOT WASTE TANK
TO SEWER
SECONDARY BLOWDOWN
V.C. COOLING WATER OUTLET
MONITOR
PB
2000 PSI
125 PSI
PRESSURE ALARM
TO FILTER INLET HEADER
V.C. COOLING WATER INLET
PRIMARY BLOWDOWN COOLER
DEMIN CONN
VENT TO COMPR. INLET HDR
O2 ANALYZER
CW
H2 ANALYZER
DRAIN
PRIMARY BLOWDOWN
NOTES:
FOR GENERAL NOTES, GRAPHICAL SYMBOLS & ETC., SEE F9-33-1008-1
LIST OF MATERIAL
APPLICATION
QTY REQD
R. LAMPHARAN
LINE DIAGRAMS COMPONENT PRIMARY CIRCUIT
U.S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.
F
M 11594-52
NONE
STEAM IN
2

10.0 PRIMARY SYSTEM INSTRUMENTATION AND WIRING

10.1 INSTRUMENTATION

Primary system instrumentation has been designed for as close an approach to 100 percent availability as is technically and economically feasible.

10.1.1 Flow

Three primary coolant flow measurements will be made by differential pressure cells, which will then send an electrical signal to a circuit. The circuit will:

(a) send as its output the highest of the three inputs from the d/p cells to the primary coolant flow indicator and to the reactor thermal power computer;

(b) monitor the deviation between the three inputs and actuate an instrument failure alarm if the deviation exceeds a pre-set value;

(c) actuate a low flow alarm and a low flow scram if at least two of the three inputs fall below the corresponding pre-set alarm and scram values.

Due to the peculiarities of the piping arrangement, no primary flow element (e.g., orifice plate, Gentile tube) can be used. The flow will be measured as a function of the differential pressure across the primary coolant pump, the reactor, and the steam generator. Probes will be inserted into the piping upstream and downstream of these items and will lead to the differential pressure cells.

10.1.2 Temperature

Temperature in the primary coolant loop will be measured at six points - three in the hot leg and three in the cold leg. The electrical signals from the temperature sensors will be sent to circuitry which will:

(a) average the signals and send the result to the average temperature recorder;

(b) select the highest of the three hot leg signals and send a corresponding signal to the reactor hot leg temperature indicator;

(c) select the highest of the three matched pair, hot leg - cold leg differential temperature signals and send a corresponding signal to the primary coolant differential temperature indicator and to the reactor thermal power computer;

(d) actuate a high temperature alarm and a high temperature scram if at least two of the three hot leg signals exceed the pre-set alarm and scram values;

(e) monitor the deviation between the differential temperature inputs and actuate an instrument failure alarm if the deviation exceeds a pre-set value.

The temperature of the pressurizer steam blanket will be measured and indicated on the console, as will be the pump bearing temperatures. High temperature of the latter will be alarmed. Three resistance bulbs will measure the temperature of the upper shield tank; three more, the temperature of the lower shield tank, another three, the temperature of the spent fuel tank, and another one the temperature of the vapor container. These latter ten will be read out on one console-mounted indicator by means of a selector switch. Local thermometers measure the temperatures at the inlet and outlet of the cooler in the spent fuel recirculation line.

10.1.3 Pressure

Primary coolant pressure will be measured at the static leg of the pressurizer. Three pressure measurements will be taken and electrical signals sent to circuitry which will:

(a) select the highest of the three signals and send it to the primary coolant pressure indicator, to the primary coolant pressure controller, and to the primary coolant rod drive seal water differential pressure controller;

(b) actuate high and low pressure alarms and scrams if at least two of the three inputs fall outside pre-set values;

(c) monitor the deviation between the three inputs and actuate an instrument failure alarm if the deviation exceeds a pre-set value.

The primary coolant pressure controller sends its output signal to the pressurizer heaters. Vapor container pressure is measured and indicated at the console, and high pressure is alarmed. Pressure downstream of the shield tank recirculating pump is measured and indicated at the console. High pressure downstream of the control rod seals and downstream of the spent fuel tank recirculation pump is sensed by local pressure switches and alarmed at the console. A pressure gauge measures the pressure downstream of the filter in the spent fuel recirculation line.

10.1.4 Level

Pressurizer level is measured by two differential pressure cells. One sends a signal to console indication and high and low level alarms, and the other sends a signal to a controller which operates a motor-operated valve in the primary makeup line to the primary coolant loop.

Steam generator shell side level is similarly measured by two differential pressure cells, one sending a signal to console indication and high and low alarms; the other sending a signal to a controller which is part of the three-element feedwater control.

Local level switches sound high and low alarms on the upper shield tank, lower shield tank and spent fuel tank.

10.1.5 Reactor Thermal Power

The reactor thermal power computer receives a differential temperature signal from the temperature circuitry and a flow signal from the flow circuitry. Thermal power is computed and a signal is sent to the reactor thermal power recorder.

10.1.6 Miscellaneous

A standby heating element is located inside the spent fuel tank for use if the normal recirculation heater is inoperable.

10.2 AUXILIARY SYSTEM INSTRUMENTATION

10.2.1 Primary Blowdown

Primary blowdown from the primary blowdown cooler passes through a pressure reducing valve which has pressure gauges located upstream and downstream of it. A pressure switch downstream of the pressure reducing valve alarms on high pressure and another switch, on still higher pressure, shuts a solenoid valve upstream of the pressure reducing valve.

A local thermometer measures the temperature of the primary blowdown and a flow meter measures its flow. A pressure regulating valve maintains the pressure.

The rod drive seal leakoff line to the initial filter inlet header is monitored for temperature by a thermometer, and for pressure by a pressure switch which sounds an alarm when the pressure exceeds a pre-set value. Pressure gauges are located upstream of the initial filters, upstream and downstream of the demineralizers, and downstream of the final filters. Differential pressure switches located across the demineralizers and both sets of filters sound an alarm on high differential pressure.

Conductivity is measured downstream of the final filters and is recorded at the console. High conductivity is alarmed. Flow is indicated locally.

pH is measured upstream of the initial filters and upstream of the primary makeup tank, and is indicated at the console. High and low pH are alarmed.

Oxygen content is analyzed upstream of filter inlet and is recorded at the console.

Undissolved hydrogen is monitored downstream of the oxygen analyzer. It is indicated at the console. Excessive undissolved hydrogen is alarmed. Still greater excess causes the vapor container to be vented. Even greater excess scrams the reactor.

10.2.2 Primary Makeup

Primary makeup tank level is locally indicated by a gauge glass. The level in the tank is controlled by means of a float operated valve controlling the flow of fresh water to the filter inlet header of the primary blowdown system. High and low levels are alarmed by means of level switches. A pressure gauge indicates the tank's pressure and a pressure switch actuates an alarm upon high pressure.

A flow switch actuates an alarm upon low flow in the suction header of the primary makeup pumps. Pressure gauges are located in the discharge lines of the primary makeup pumps.

Pressure in the primary makeup lines to the vapor container is kept constant by a back-pressure regulating valve in a bypass line to the primary makeup tank. The valve maintains pressure by allowing more or less makeup water to return to the tank. A pressure regulating valve reduces the pressure of the water returning to the tank to match that in the tank.

The pressure of the makeup water to the control rod drive seals must be maintained slightly above that of the primary coolant. This is accomplished as follows: The pressure of the makeup water to the seals is sensed and a signal is transmitted to a console-mounted differential pressure indicator-controller. The controller also receives a signal from the primary coolant pressure circuitry. The pressure difference is computed and maintained at a preset value, by a throttling signal to one of two control valves arranged in parallel located in the makeup water line to the seals. If the pressure difference falls below a pre-set limit, an alarm will be sounded.

10.2.3 Vapor Container Cooling Water System

This system supplies cooling water for the various heat exchange apparatus and equipment requiring cooling water. A pressure gauge is located at the discharge of each of the two cooling water pumps. A flow switch in the discharge header of the pumps sounds an alarm when the cooling water flow falls below a pre-set value. Just before the alarm is sounded the flow switch will have caused a pump switchover.

The cooling water line from the primary coolant pump has its temperature indicated by a thermometer and its flow indicated by a local flow meter.

The main cooling water line is locally monitored for temperature and flow by a thermometer and a flow meter before entering the cooling water cooler.

The vapor container cooling water storage tank has a gauge glass indicating the liquid level; a low level alarm is sounded by a float actuated switch; and another float switch causes the cooling water pumps to shut down on extremely low level. A float-operated valve controls the level in the storage tank by regulating makeup water from the cold condensate storage tank.

The vapor container cooling water tank has a thermometer measuring the liquid temperature.

10.2.4 Secondary Blowdown

Secondary blowdown is monitored for temperature by two thermometers and flow by a local flow meter. Pressure gauges measure the pressure upstream and downstream of a pressure reducing valve. A conductivity cell sends its signal to a console mounted recorder with a high conductivity alarm contact. High radiation in the secondary blowdown causes a three-way valve to dump the blowdown liquid to the hot waste system.

10.2.5 Hot Waste System

A pressure transmitter monitors the pressure in each of the three hot waste tanks and in the discharge line of each of the pumps in the system. The pressures are indicated at the console on one indicator with a selector switch, except for the evaporator vacuum pump and the waste gas compressor, which have individual console mounted indicators.

The pressure drop across each of the two filters is monitored by a differential pressure switch which sounds an alarm when the pressure drop exceeds a pre-set value.

The pressure in the waste gas storage cylinder manifold is measured by a pressure transmitter and is indicated on the console multi-position indicator described above.

Each of the three hot waste tanks has its level measured by a level transmitter. These three transmitters have their signals read out at the console on one indicator with a selector switch. In addition, float-actuated level switches sound alarms on high and low level in each of the three tanks.

The 200 gal holding tank and the 300 gal condensate tank have high and low level alarm switches (float actuated) and float actuated switches which start and stop their respective discharge pumps (Dwg M 11594-48).

10.2.6 Instrument List

Item	Function	Location	Quantity
Pressure indicator PI-263	H_2 flask	Local	1
Pressure indicator PI-253	Primary makeup tank	Local	1
Pressure indicator PI-258	Primary makeup pump discharge	Local	1
Pressure indicator PI-259	Spare primary makeup pump discharge	Local	1
Pressure indicator PI-211	Input to first primary blowdown filter	Local	1
Pressure indicator PI-213	Input to demineralizers	Local	1
Pressure indicator PI-215	Input to final filters	Local	1
Pressure indicator PI-217	Downstream of filters and demineralizers	Local	1
Pressure indicator PI-360	Spent fuel tank recirc. filter discharge	Local	1
Pressure indicator PI-202	Primary blowdown cooler discharge	Local	1
Pressure indicator PI-203	Downstream of primary blowdown pressure packing valve	Local	1
Pressure transmitter PI-160, 161, 162	Pressurizer (three elements for pressurizer)	Pressure monitoring panel (control room readouts)	3
Pressure transmitter PX-353	Shield tank recirc. pump filter discharge	Local (control room readout)	1
Pressure indicator PI-160	Pressurizer	Control room	1
Pressure indicator Controller PIC-161	Pressurizer Pressure control	Control room	1

Item	Function	Location	Quantity
Switch HPSA-160	Primary coolant system-high pressure alarm	Control room	1
Switch LPSA-160	Primary coolant system low pressure alarm	Control room	1
Switch HPSS-160	Primary coolant system high pressure scram	Control room	1
Switch LPSS-160	Primary coolant system low pressure scram	Control room	1
Pressure switch HPSA-205	Primary blowdown downstream of press. red. valve high press. alarm	Local control room annunciator	1
Pressure indicator PI-353	Shield tank pump filter	Local	1
Pressure switch HPSC-205	Pri. blowdown downstream of press. red. valve-high press. trip of solenoid valve located downstream of primary blowdown cooler	Local	1
Pressure switch HPSA-254	Primary makeup tank high pressure alarm	Local (control room annunciator)	1 1
Pressure switch HPSA-356	Seal leakoff to filters - high pressure alarm	Local (control room annunciator)	1
Pressure switch HPSA-359	Spent fuel tank recirc. pump discharge high pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-212	Across primary blowdown filters upstream of demineralizer - high diff. pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-214	Across demineralizers high differential pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-216	Across pri. blowdown filters downstream of demineralizers - high diff. pressure alarm	Local (control room annunciator)	1
Pressure transmitter PX-362	Vapor container	Pressure monitoring panel	1
Pressure indicator PI-362	Vapor container	Control room	1

Item	Function	Location	Quantity
Switch HPSA-160	Primary coolant system-high pressure alarm	Control room	1
Switch LPSA-160	Primary coolant system low pressure alarm	Control room	1
Switch HPSS-160	Primary coolant system high pressure scram	Control room	1
Switch LPSS-160	Primary coolant system low pressure scram	Control room	1
Pressure switch HPSA-205	Primary blowdown downstream of press. red. valve high press. alarm	Local control room annunciator	1
Pressure indicator PI-353	Shield tank pump filter	Local	1
Pressure switch HPSC-205	Pri. blowdown downstream of press. red. valve-high press. trip of solenoid valve located downstream of primary blowdown cooler	Local	1
Pressure switch HPSA-254	Primary makeup tank high pressure alarm	Local (control room annunciator)	1 1
Pressure switch HPSA-356	Seal leakoff to filters - high pressure alarm	Local (control room annunciator)	1
Pressure switch HPSA-359	Spent fuel tank recirc. pump discharge high pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-212	Across primary blowdown filters upstream of demineralizer - high diff. pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-214	Across demineralizers high differential pressure alarm	Local (control room annunciator)	1
Differential pressure switch HDPSA-216	Across pri. blowdown filters downstream of demineralizers - high diff. pressure alarm	Local (control room annunciator)	1
Pressure transmitter PX-362	Vapor container	Pressure monitoring panel	1
Pressure indicator PI-362	Vapor container	Control room	1

Item	Function	Location	Quantity
Switch HPSA-362	Vapor container high pressure alarm	Control room	1
Pressure reducing valve PCV-204	Downstream of primary coolant blowdown cooler	Local	1
Pressure regulating valve PCV-208	Primary cooler filter inlet	Local	1
Pressure transmitter PX-262	Primary makeup to rod drive seals	Local	1
Diff. press. ind. control DPIC-262	Primary cool. - rod drive seal water	Console	1
Switch LDPSA-262	Primary cool. - rod drive seal diff. pressure alarm	Console	1
Control valve DPCV-262	Primary cool. -rod drive seal water	Local	2
Pressure regulating valve PCV-260	Primary makeup water	Local	1
Pressure regulating valve PCV-261	Primary makeup return	Local	1
Pressure indicator PI-264	Hydrogen to primary makeup tank	Local	1
Pressure regulating valve PCV-265	Hydrogen to primary makeup tank	Local	1
Pressure indicator PI-266	Hydrogen to primary makeup tank	Local	1
Pressure transmitter PX-910	Cond. tank pump discharge	Local	1
Pressure transmitter PX-912	Waste gas comp. discharge	Local	1
Pressure transmitter PX-913	Waste gas manifold	Local	1
Pressure indicator PI-901 with selector switch	H.W. system pressures	Console	1
Diff. pressure switch HDPSA-911	H.W. outlet filter	Local	1
Diff. pressure switch HDPSA-909	H.W. inlet filter	Local	1
Pressure transmitter PX-901	H.W. tank no. 1	Local	1
Pressure transmitter PX-902	H.W. tank no. 1 pump discharge	Local	1
Pressure transmitter PX-903	H.W. tank no. 2	Local	1

Item	Function	Location	Quantity
Pressure transmitter PX-904	H.W. tank no.2 pump discharge	Local	1
Pressure transmitter PX-905	H.W. tank no. 3	Local	1
Pressure transmitter PX-906	H.W. tank no. 3 pump discharge	Local	1
Pressure transmitter PX-907	Evap. cond. input	Local	1
Pressure transmitter PX-908	Evap. vac. pump discharge	Local	1
Pressure indicator PI-380	V.C. cooling water pump discharge	Local	2
Pressure indicator PI-602	Secondary blowdown	Local	1
Pressure reducing valve PCV-604	Secondary blowdown	Local	1
Pressure indicator PI-603	Downstream of PCV-604	Local	1
Pressure indicator PI-908	Evap. vacuum pump discharge	Console	1
Pressure indicator PI-912	H.W. gas compressor discharge	Console	1
Temperature element TE-154	Steam generator outlet-tube side	Control room readout	1
Temperature element TE-155	Steam generator inlet-tube side	Local (control room readout)	1
Temperature element TE-156	Primary coolant pump suction	Local (control room readout)	1
Temperature element TE-157	Reactor outlet	Local (control room readout)	1
Temperature element TE-158	Reactor outlet	Local (control room readout)	1
Temperature element TE-159	Primary coolant pump discharge	Local (control room readout)	1
Temperature element TE-164	Pressurizer	Local (control room readout)	1
Temperature element TE-364	Primary coolant pump upper thrust bearing	Local (control room readout)	1
Temperature element TE-365	Primary coolant pump lower thrust bearing	Local (control room readout)	1
Temperature indicator TI-206	Pri. blowdown downstream of pressure reducing valve	Local	1
Temperature indicator TI-357	Seal leakoff of filters	Local	1

Item	Function	Location	Quantity
Temperature indicator TI-158	Reactor outlet	Control room	1
Temperature indicator TI-164	Pressurizer	Control room	1
Diff. temp. indicator DTI-155	Differential temperature across reactor	Control room	1
Average temp. recorder ATR-154	Primary coolant average temperature	Control room	1
Temperature switch HTSA-150	Primary coolant high temperature alarm	Control room	1
Temperature switch HTSC-150	Primary coolant high temperature scram	Control room	1
Temperature switch HTS-364	P. C. pump-upper thrust brg. high temp. alarm	Control room	1
Temperature switch HTS-365	P. C. pump-lower thrust brg. high temp. alarm	Control room	1
Temperature indicator TI-364	Primary coolant pump pump bearings	Control room	1
Temperature indicator TI-361	Spent fuel tank recirc. cooler-heater inlet	Local	1
Temperature indicator TI-375	Spent fuel tank recirc. cooler-heater disch.	Local	1
Temperature elements TE-366, 367, 368	Upper shield tank	Local (control room readout)	3
Temperature elements TE-369, 370, 371	Lower shield tank	Local (control room readout)	3
Temperature elements TE-372, 373, 374	Spent fuel tank	Local (control room readout)	3
Temperature element TE-363	Vapor container	Local (control room readout)	1
Temp. indicator and selector switch TI-363	Miscellaneous temperatures	Control room	1
Temperature indicator TI-379	V. C. cooling H_2O tank	Local	1
Temperature indicator TI-382	C. W. from P. C. pump	Local	1
Temperature indicator TI-384	C. W. to C. W. cooler	Local	1
Temperature indicator TI-601	Secondary blowdown	Local	1
Temperature indicator TI-606	Secondary blowdown	Local	1
Flow indicator FI-219	Input to primary make-up tank	Local	1

Item	Function	Location	Quantity
Flow transmitter FX-151, 152, 153	Primary coolant	Local to remote readout	3
Flow indicator FI-151	Primary coolant	Control room	1
Flow switch LFSA-150	Low flow alarm primary coolant	Control room	1
Flow switch LFSS-150	Low flow scram primary coolant	Control room	1
Flow switch LFSA-257	Low flow alarm primary makeup tank outlet	Local alarm in control room	1
Flow indicator FI-207	Primary blowdown to filters and demineralizers	Local	1
Flow meter FI-383	C. W. from P. C. pump	Local	1
Flow meter FI-385	C. W. to C. W. cooler	Local	1
Flow switch LFSA-381	V. C. C. W. pumps low flow alarm	Local	1
Flow switch LFSC-381	V. C. C. W. pump low flow switchover	Local	1
Flow meter FI-608	Secondary blowdown	Local	1
Level elements LE-165, 166	Pressurizer level	Local (control room readout)	2
Level elements LE-401, 402	Steam generator level	Local (control room readout)	2
Level indicating controller LIC-166	Pressurizer level	Control room	1
Level indicator LI-165	Pressurizer level	Control room	1
Level indicating controller LIC-402	Steam generator level	Control room	1
Level indicator LI-401	Steam generator level	Control room	1
Level gauge LG-251	Primary makeup tank level	Local	1
Level switch HLSA-165	Pressurizer high level alarm	Control room	1
Level switch LLSA-165	Pressurizer low level alarm	Control room	1
Level switch HLSA-401	Steam generator high level alarm	Control room	1
Level switch LLSA-401	Steam generator low level alarm	Control room	1

Item	Function	Location	Quantity
Level switch HLSA/LLSA-351	Upper shield tank high and low level alarm	Local (control room annunciator)	1
Level switch HLSA/LLSA-352	Lower shield tank high and low level alarm	Local (control room annunciator)	1
Level switch HLSA/LLSA-358	Spent fuel tank high and low level alarm	Local (control room annunciator)	1
Level switch HLSA-255	Primary makeup tank high level alarm	Local	1
Level switch LLSA-256	Primary makeup tank low level alarm	Local (control room annunciator)	1
Level controller LC/LCV-252	Primary makeup tank	Local	1
Level element LE-914	H. W. tank no. 1	Local	1
Level indicator LI-914	H. W. tank no. 1, 2, 3	Console	1
Level switch HLSA-915	H. W. tank no. 1	Local	1
Level switch LLSA-915	H. W. tank no. 1	Local	1
Level element LE-916	H. W. tank no. 2	Local	1
Level switch HLSA-917	H. W. tank no. 2	Local	1
Level switch LLSA-917	H. W. tank no. 2	Local	1
Level element LE-918	H. W. tank no. 3	Local	1
Level switch HLSA-919	H. W. tank no. 3	Local	1
Level switch LLSA-919	H. W. tank no. 3	Local	1
Level switch HLSA-920	H. W. condensate tank	Local	1
Level switch HLSC-920	H. W. condensate tank pump	Local	1
Level switch LLSA-921	H. W. condensate tank	Local	1
Level switch LLSC-921	H. W. condensate tank pump	Local	1
Level indicator LI-376	V. C. C. W. tank	Local	1
Level control and valve LCV-377	V. C. C. W. tank	Local	1

Item	Function	Location	Quantity
Level switch LLSC-378	V.C. C.W. tank pump shutdown	Local	1
Level switch LLSA-378	V.C. C.W. tank low level alarm	Local	1
Reactor thermal power computer and recorder QR-163	Reactor thermal power power	Control room	1
Oxygen analyzer O2X-210	Determine oxygen Content of: (1) Contaminated waste to hot waste tank (2) Pri. blowdown up-stream of primary makeup tank (3) Pri. makeup pump suction header (4) Secondary blowdown (5) Condenser hotwell	Local (control room readout)	1
Oxygen recorder O2R-210	Oxygen content	Control room	1
Conductivity element CE-218	Pri. blowdown upstream of primary makeup tank	Local (control room readout)	1
Conductivity recorder CR-218	Conductivity of various streams in primary and secondary systems	Control room	1
Switch HCSA-218	Pri. blowdown to primary m.u. tank high conductivity alarm	Control room	1
pH Detector pHX-220	Pri. blowdown to primary makeup tank	Local	1
pH Indicator pHI-209 with selector switch	Primary blowdown to filters and makeup tank after pHI-209	Control room	1
Switch HPHSA-220	Pri. blowdown to pri. makeup tank high pH alarm	Console	1
Switch LPHSA-220	Pri. blowdown to pri. makeup tank low pH alarm	Console	1
Transmitter pHX-209	Pri. blowdown to filters	Local	1
Switch HpHSA-209	Pri. blowdown to filters	Console	1
Switch LpHSA-209	Pri. blowdown to filters	Console	1

Item	Function	Location	Quantity
Control valve LCV-166	Pri. makeup to P.C.	Local	1
Conductivity cell CE-607	Secondary blowdown	Local	1
Switch HCSA-607	Secondary blowdown	Console	1
Control valve RCV-605	Secondary blowdown high rad. dump	Local	1
Level switch HLSA/HLSC-922	H.W. 200 gal. holding tank	Local	1
Level switch HLSA/LLSA-923	H.W. 200 gal. holding tank	Local	1

10.3 VAPOR CONTAINER WIRING

The vapor container wiring design will follow the same general intent of simple erection and dismantling. However, special problems are present since all elements contained inside the vapor container are not mounted on the primary skid.

When shipped, the primary skid will contain the reactor, steam generator and primary coolant pump. Instruments mounted on these elements will have leads carried to a junction box mounted on the vapor container wall, through rigid conduit installed on erection. Instrument leads from other elements installed within the vapor container will be carried to the same junction box in the same manner. From this junction box, leads will be run in multi-conductor cables contained in rigid conduit to and through the vapor container penetrations to a receptacle termination on the outside of the vapor container. This entire run will be waterproof for the protection of the conductors.

The power leads for the primary coolant pump motor will be run through waterproof rigid conduit to a junction box mounted on the pump. This junction box will contain receptacles where the incoming leads can be plugged in to serve the motor. There will be a short run in flexible, water tight conduit to make unplugging simple and to carry the power lead up to rigid conduit on the vapor container wall. This conduit will carry the leads out through the penetration to a receptacle termination on the outside of the vapor container. All of this run will be waterproof to maintain the integrity of the conductor and the vapor container.

The control rod drives will be shipped separately from the primary skid and installed at the site. The leads for these drives will be brought through flexible, water tight conduit to a junction box mounted on the concrete shield near these drives. From this point multi-conductor plug-in cable will carry the leads through flexible and rigid conduit to the plug in receptacle termination on the outside of the vapor container.

The terminal end of the pressurizer heaters will be exposed in a terminal box on the pressurizer. The heater leads will be carried from this box through flexible water tight conduit to a junction box mounted on the vapor container wall and from this point through a multi-conductor cable contained in conduit to the receptacle termination on the outside of the vapor container.

Leads from the radiation monitors and telephone circuits will be carried to a common junction box on the vapor container wall and out to the receptacle termination on the outside of the vapor container in a fashion similar to the above runs. Lighting leads will be entirely in rigid conduit from the lights to the junction box and out.

Wiring the vapor container in the above fashion will simplify site erection and keep vapor container penetrations to a minimum. At the receptacle termination on the outside of the vapor container multi-conductor cables could be plugged in and trained up into the open overhead raceways which carry the conductors throughout the plant.

The details of the vapor container penetrations are shown on Dwg. M 11594-74.

10.4 WIRING, PRIMARY AUXILIARIES SKID

The motor controls for the motors located on the primary auxiliaries skid will be mounted on the skid with the motors. Power leads from the motors will be run to the motor starters through rigid conduit permanently installed on the skid. The power leads will leave these motor starters through flexible cable which will be trained up to the open raceway which will carry power and control wiring through the station.

Control and instrument wiring will be carried to a junction box mounted at the end of the skid and collected on multi-conductor receptacles. Multi-conductor cables will be plugged into these receptacles and then trained up into the open cable raceway to be carried to other points in the station.

10.4.1 Interconnecting Wiring and Piping

Piping from the primary skid to the auxiliary skid with its problems of vapor container penetrations, is a field erection aspect. The geometry of lines operating at elevated temperatures is such as to allow anchoring to the concrete shield-wall base while maintaining the pressure integrity of the vapor container. The penetrations are made below the floor level for the attenuation of neutron streaming. Shut-off valves and drips are provided in the pit but are accessible for setting and maintenance without descending into the pit. Piping is supported from catwalk framing inside the vapor container and from the overhead outside it. Piping to and from the spent fuel pit is entirely 304 SS buried, as is piping to the contaminated waste facilities and storage.

The vapor container and primary auxiliaries skid wiring is described in preceding paragraphs. On all other skids the wiring contained within the skid is installed in rigid and flexible conduit.

Control wiring leaving the skid is brought to junction boxes and terminated. From this point multi-conductor plug-in cables carry the wiring to the skid designated in overhead raceways. At each skid connection, a short run of flexible conduit, for the purpose of disconnection, and rigid conduit, protect the cable between the raceway and skid.

Power leads are run from point to point in the station in raceways. At the skids, rigid conduit carries the power leads directly to the equipment control to which it is connected.

10.5 ELECTRICAL SYSTEM

The electrical system functions to supply the station auxiliaries with their required power and to deliver the net output of the station generator to the local distribution system.

The output of the generator is delivered to the station bus at 4160 volts, 3 phase, 4 wire and 60 cycle through a 2000 amp circuit breaker. A relay protective scheme provides the generator with differential, overcurrent, and short circuit protection. Generator excitation is provided by a 40 KW motor driven exciter and a spare exciter is supplied in the event of exciter failure. A metering scheme is provided to monitor the generator output and a synchroscope is provided for paralleling operations.

Duplicate 1000 KVA transformers are connected to the 4160 volt station bus through 1200 amp circuit breakers to provide parallel service to the station auxiliaries. A metering scheme monitors the amount of load carried by each transformer and an overcurrent relaying scheme is provided to operate high and low side breakers for transformer protection.

Normally each transformer carries approximately half of the total running load of 1205 KVA and operates self-cooled. Upon loss of either transformer the tie breaker would transfer the total load to the operative transformer which would operate forced-air cooled until normal service is restored.

The auxiliary loads are divided between busses normally operated separately to insure at least partial plant operation in the event of equipment failure. This division is detailed on the One Line Diagram.

Two 2000 amp circuit breakers connect 4160 volt outgoing feeders to the station bus. A protective relay scheme is provided to protect the feeder and station bus from faults. A metering scheme monitors the load carried by each feeder and a synchronizing scheme operates in conjunction with the generator synchroscope to parallel either feeder with the generator.

The above electrical system is detailed on Dwg. M 11594-26. Values given for motor and heater sizes are preliminary as development work on the SM-2 electrical system was terminated at the time the conceptual design of the SM-2 Complex III was initiated. At this time, the process system designs were not finalized and consequently motor and heater sizes were still fluid.

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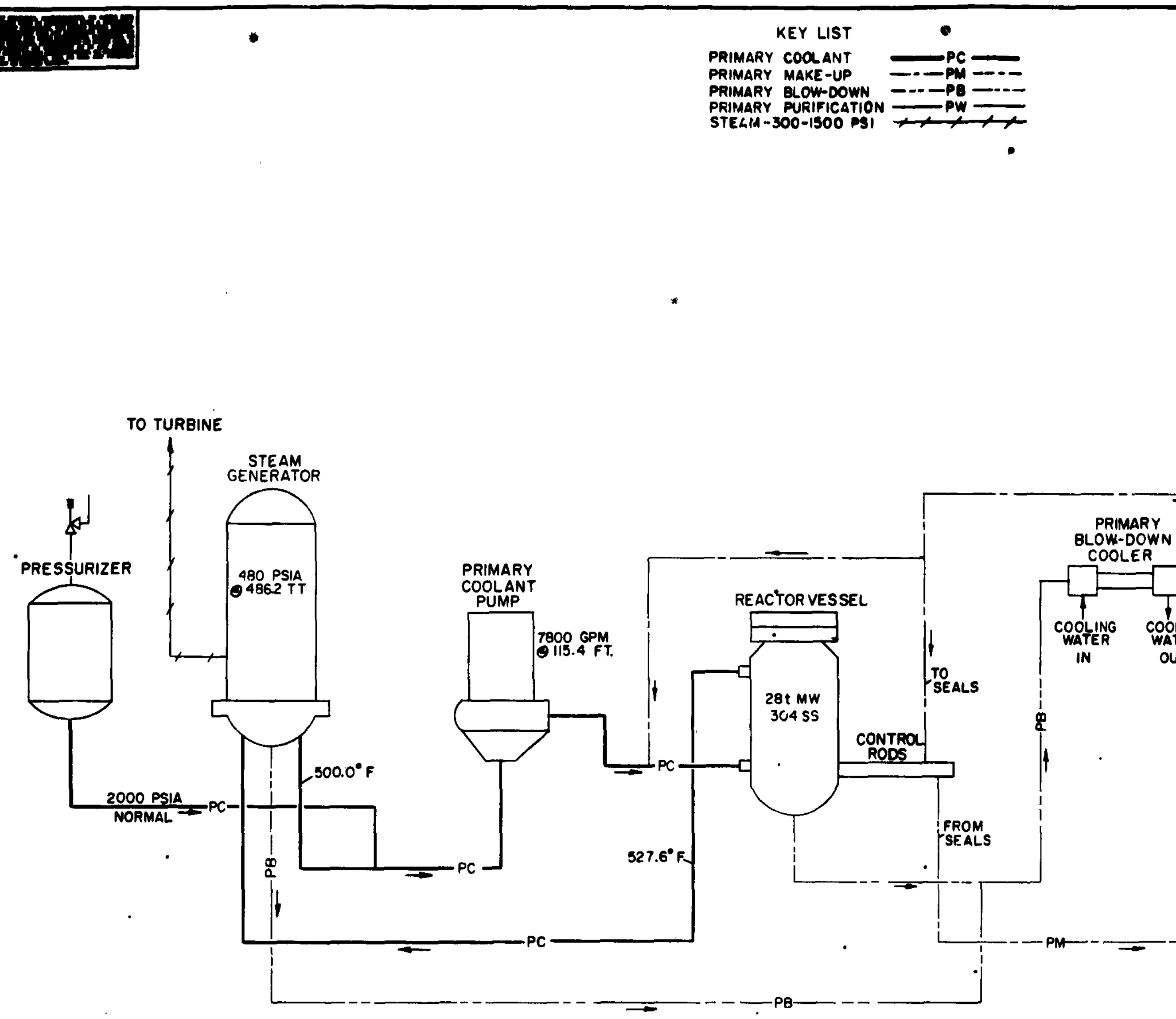
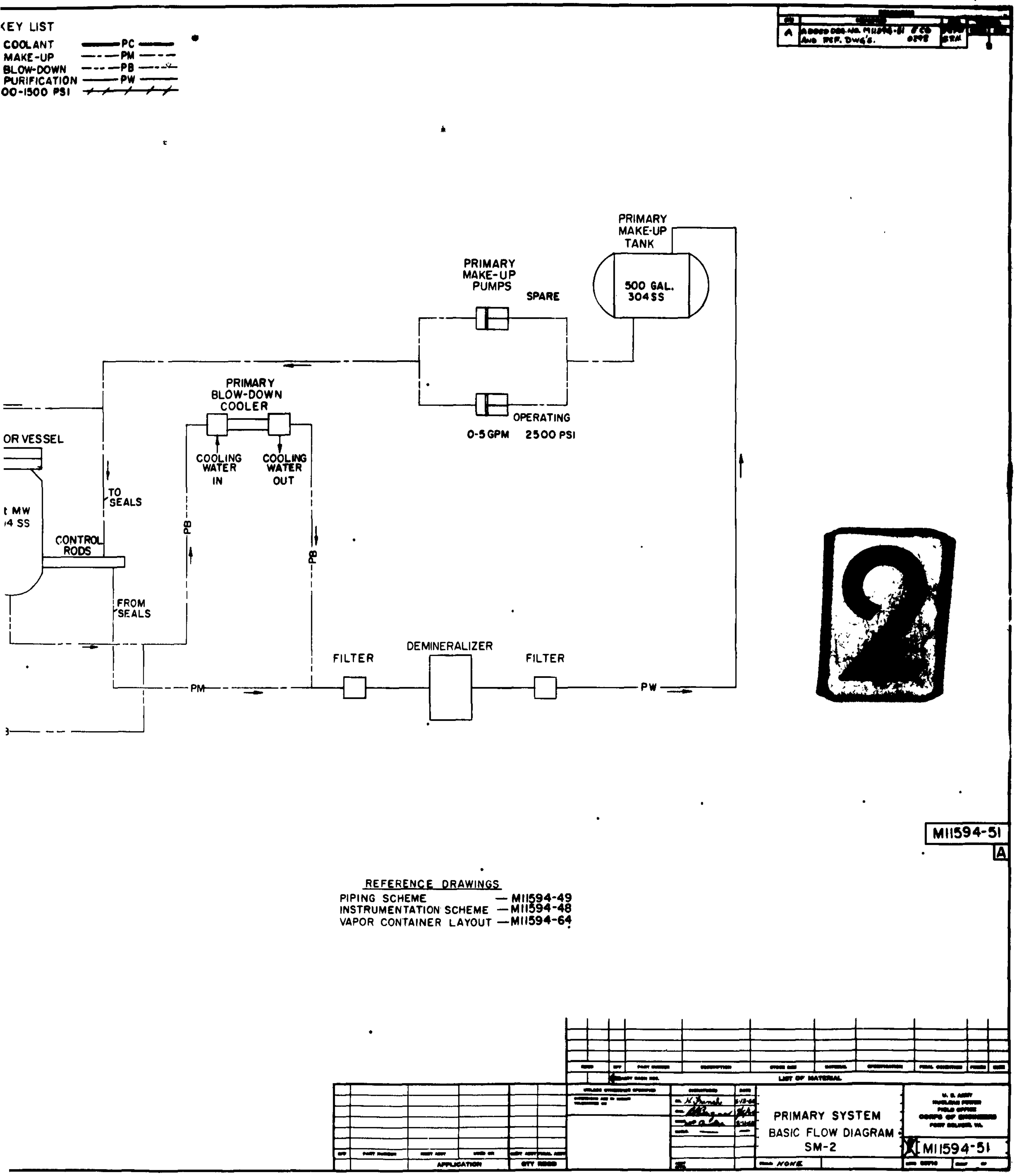


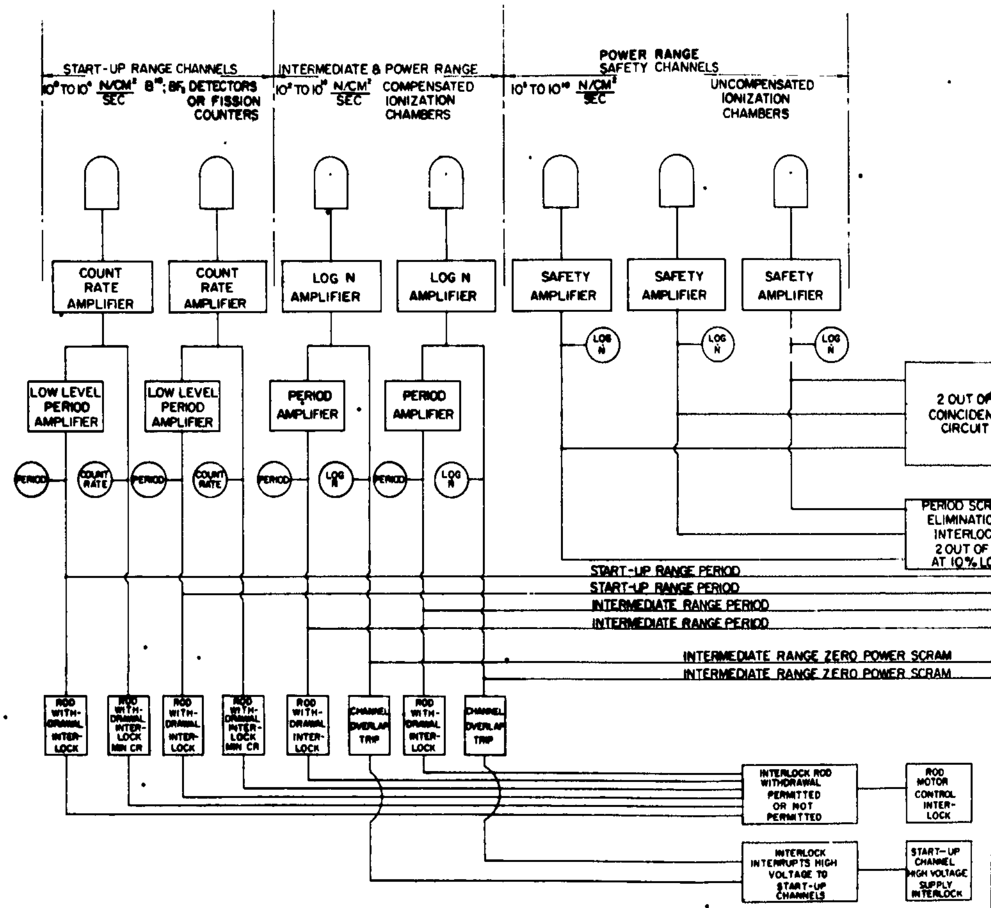
KEY LIST
PRIMARY COOLANT PC
PRIMARY MAKE-UP PM
PRIMARY BLOW-DOWN PB
PRIMARY PURIFICATION PW
STEAM-300-1500 PSI
TO TURBINE
STEAM GENERATOR
480 PSIA @ 486.2 TT
PRESSURIZER
PRIMARY COOLANT PUMP
7800 GPM @ 115.4 FT.
REACTOR VESSEL
28t MW
304 SS
CONTROL RODS
TO SEALS
FROM SEALS
PRIMARY BLOW-DOWN COOLER
COOLING WATER IN
COOL WAT OU
500.0° F
2000 PSIA NORMAL
527.6° F
PC
PB
PM

1

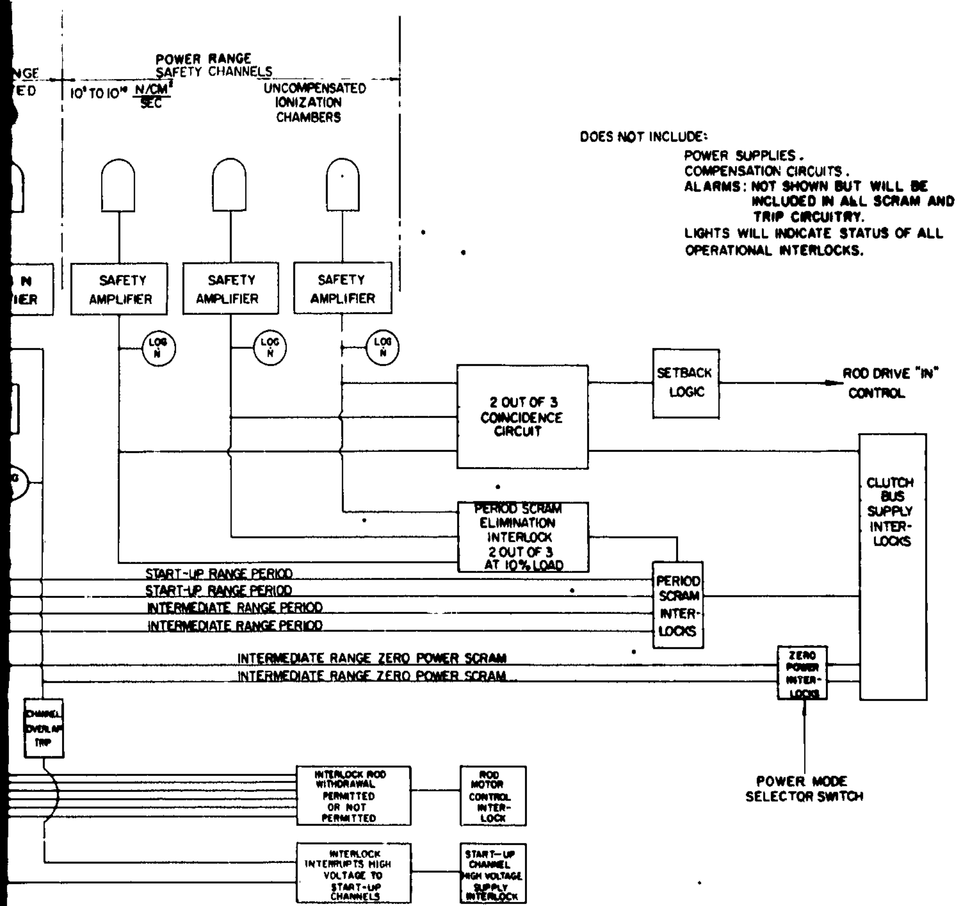


KEY LIST
COOLANT — PC
MAKE-UP — PM
BLOW-DOWN — PB
PURIFICATION — PW
00-1500 PSI
A ADDED DRG. NO. M11594-51 & CO AND REF. DWG'S.
PRIMARY MAKE-UP TANK
500 GAL. 304SS
PRIMARY MAKE-UP PUMPS
SPARE
OPERATING
0-5 GPM
2500 PSI
PRIMARY BLOW-DOWN COOLER
COOLING WATER IN
COOLING WATER OUT
OR VESSEL
t MW
4 SS
TO SEALS
CONTROL RODS
FROM SEALS
PB
PB
PM
FILTER
DEMINERALIZER
FILTER
PW
2
M11594-51
A
REFERENCE DRAWINGS
PIPING SCHEME — M11594-49
INSTRUMENTATION SCHEME — M11594-48
VAPOR CONTAINER LAYOUT — M11594-64
LIST OF MATERIAL
APPLICATION
QTY REQD
PRIMARY SYSTEM
BASIC FLOW DIAGRAM
SM-2
M11594-51
NONE

1



START-UP RANGE CHANNELS
10^0 TO 10^4 N/CM² SEC
B^{10}; BF_3 DETECTORS OR FISSION COUNTERS
INTERMEDIATE & POWER RANGE
10^2 TO 10^{10} N/CM² SEC
COMPENSATED IONIZATION CHAMBERS
POWER RANGE SAFETY CHANNELS
10^3 TO 10^{10} N/CM² SEC
UNCOMPENSATED IONIZATION CHAMBERS
COUNT RATE AMPLIFIER
COUNT RATE AMPLIFIER
LOG N AMPLIFIER
LOG N AMPLIFIER
SAFETY AMPLIFIER
SAFETY AMPLIFIER
SAFETY AMPLIFIER
LOG N
LOG N
LOG N
LOW LEVEL PERIOD AMPLIFIER
LOW LEVEL PERIOD AMPLIFIER
PERIOD AMPLIFIER
PERIOD AMPLIFIER
PERIOD
COUNT RATE
PERIOD
COUNT RATE
PERIOD
LOG N
PERIOD
LOG N
START-UP RANGE PERIOD
START-UP RANGE PERIOD
INTERMEDIATE RANGE PERIOD
INTERMEDIATE RANGE PERIOD
INTERMEDIATE RANGE ZERO POWER SCRAM
INTERMEDIATE RANGE ZERO POWER SCRAM
ROD WITH-DRAWAL INTER-LOCK
ROD WITH-DRAWAL INTER-LOCK MIN CR
ROD WITH-DRAWAL INTER-LOCK
ROD WITH-DRAWAL INTER-LOCK MIN CR
ROD WITH-DRAWAL INTER-LOCK
CHANNEL OVERLAP TRIP
ROD WITH-DRAWAL INTER-LOCK
CHANNEL OVERLAP TRIP
INTERLOCK ROD WITHDRAWAL PERMITTED OR NOT PERMITTED
ROD MOTOR CONTROL INTER-LOCK
INTERLOCK INTERRUPTS HIGH VOLTAGE TO START-UP CHANNELS
START-UP CHANNEL HIGH VOLTAGE SUPPLY INTERLOCK



DOES NOT INCLUDE:

POWER SUPPLIES.
COMPENSATION CIRCUITS.
ALARMS: NOT SHOWN BUT WILL BE INCLUDED IN ALL SCRAM AND TRIP CIRCUITRY.
LIGHTS WILL INDICATE STATUS OF ALL OPERATIONAL INTERLOCKS.

LIST OF MATERIAL

APPLICATION

QTY REQD

H FRENCH

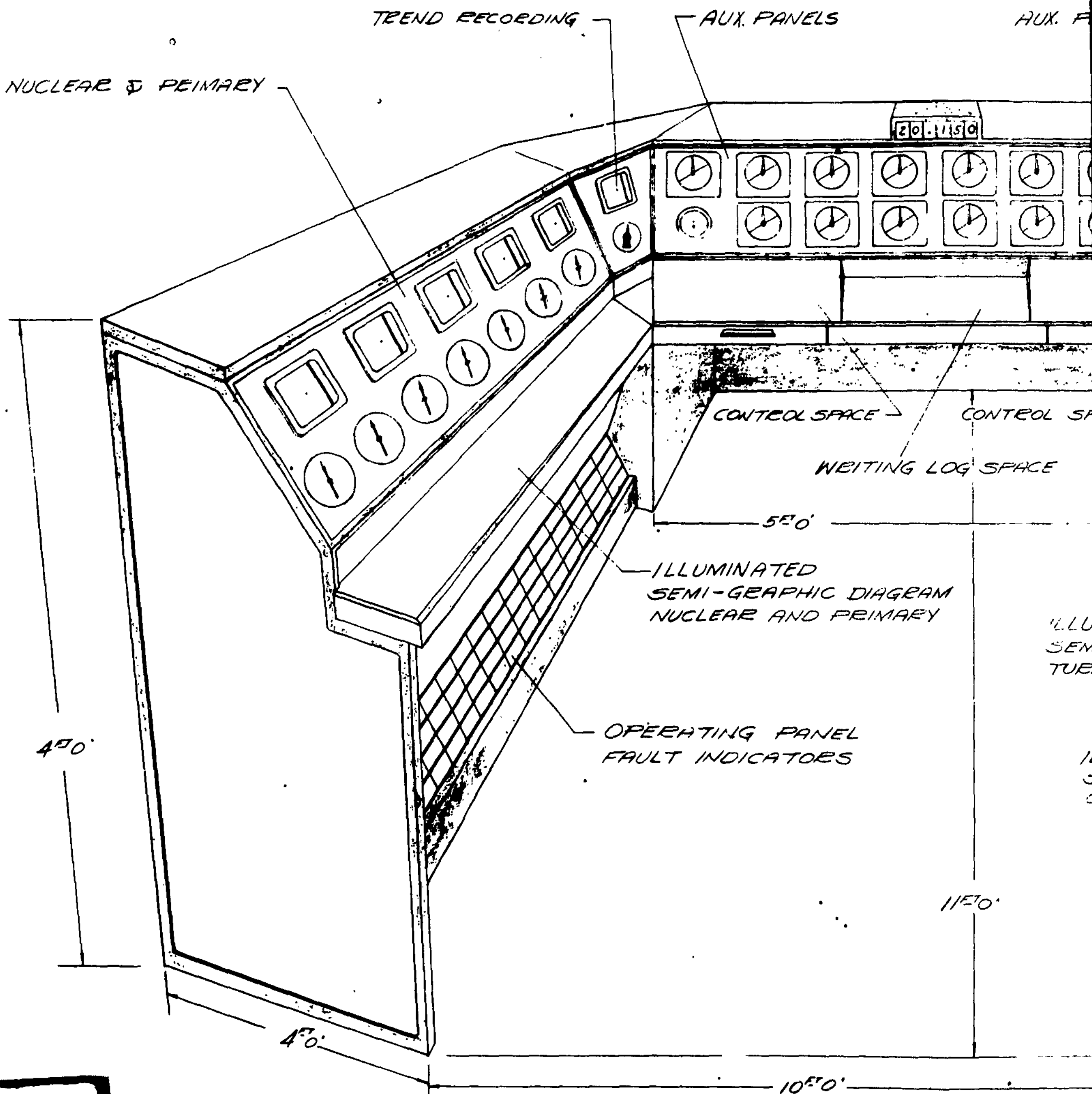
DIAGRAM
NUCLEAR INSTRUMENTATION
SM-2

U.S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F M 11594-73

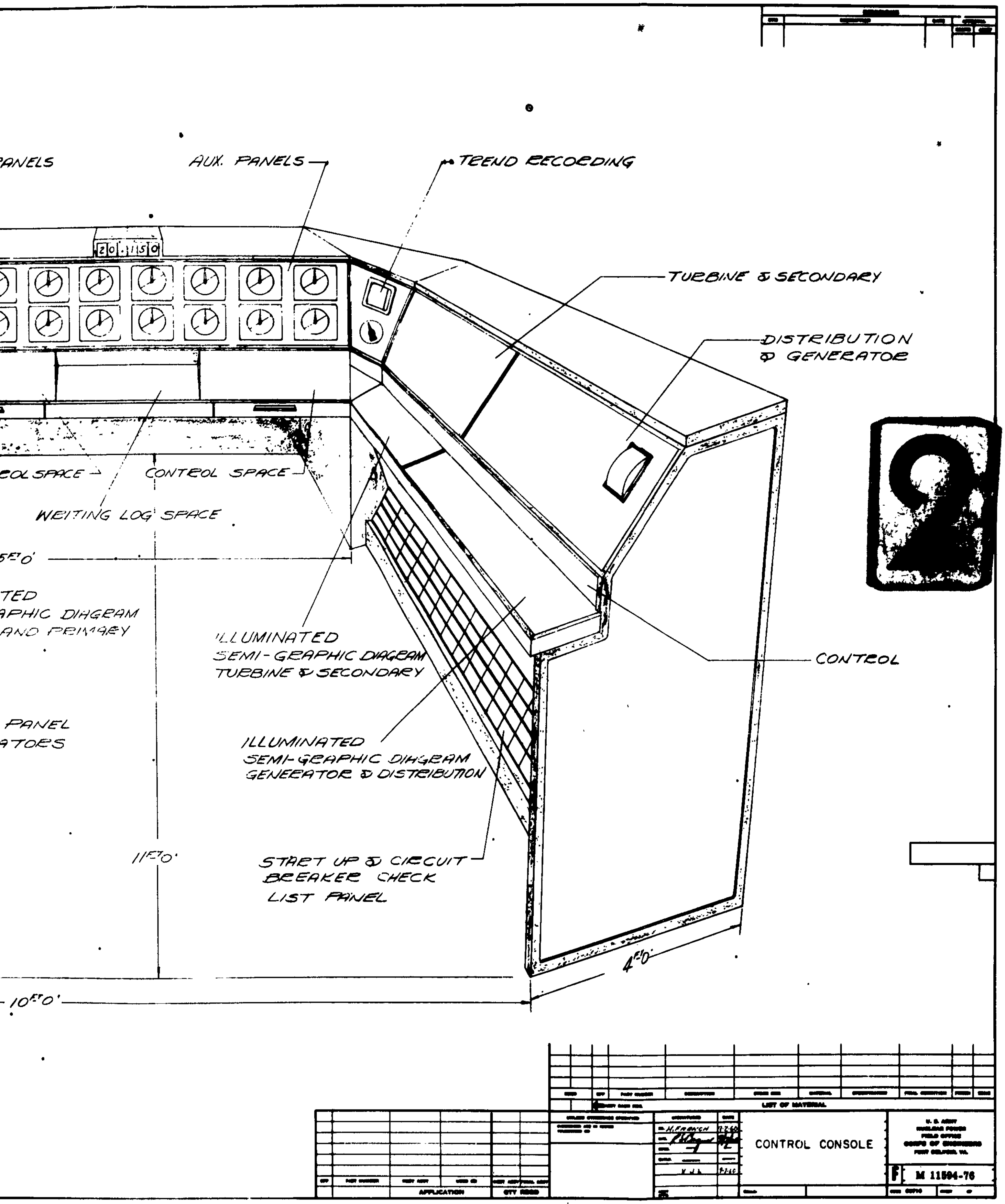
SCALE NONE

[redacted]

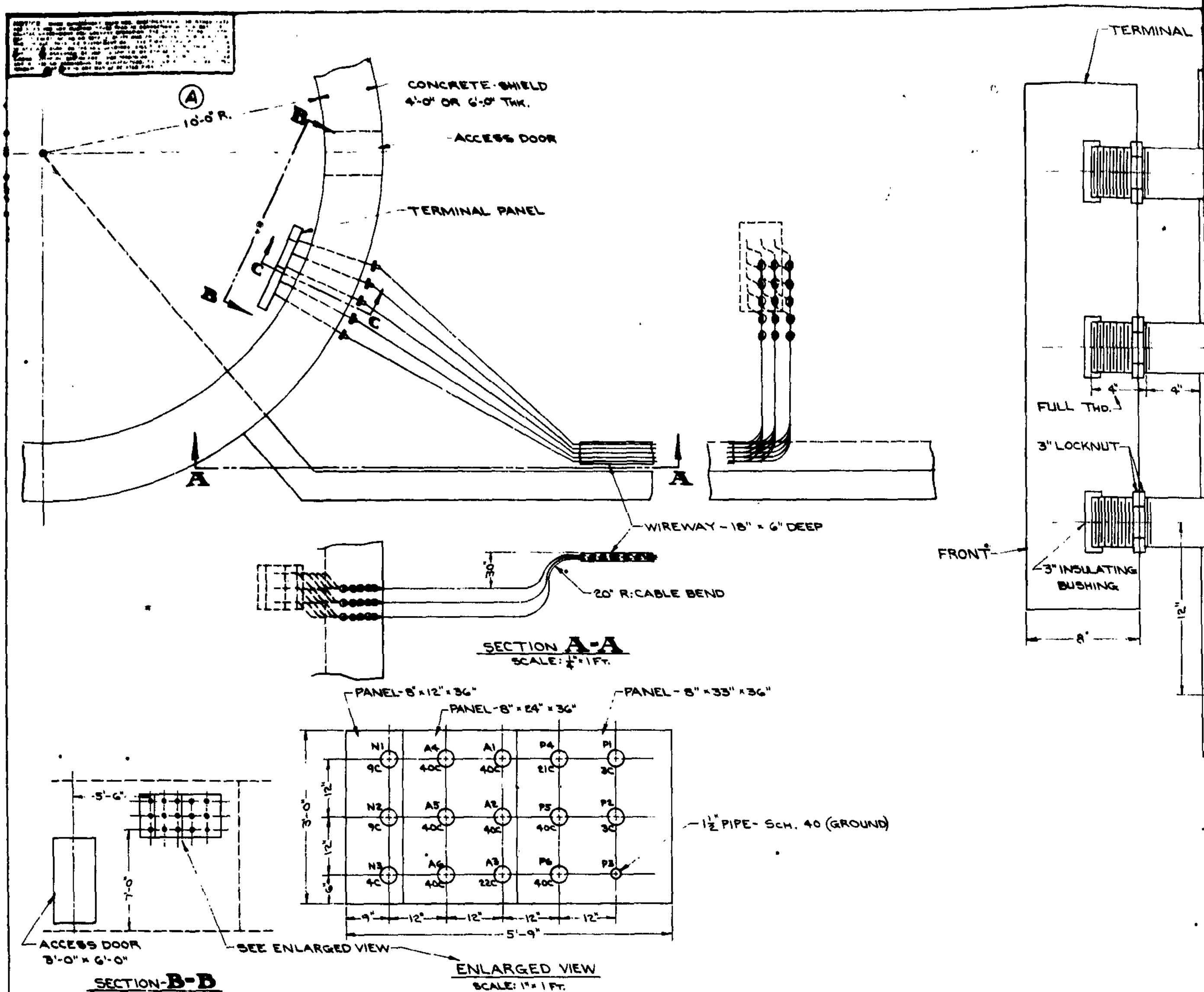


TREND RECORDING
AUX. PANELS
AUX. P
NUCLEAR & PRIMARY
CONTROL SPACE
CONTROL SP
WRITING LOG SPACE
5FT 0'
ILLUMINATED
SEMI-GRAPHIC DIAGRAM
NUCLEAR AND PRIMARY
OPERATING PANEL
FAULT INDICATORS
4FT 0'
4FT 0'
11FT 0'
10FT 0'

1



PANELS
AUX. PANELS
TREND RECORDING
TURBINE & SECONDARY
DISTRIBUTION & GENERATOR
20.1150
ROL SPACE
CONTROL SPACE
WRITING LOG SPACE
5FT 0'
TED
APHIC DIAGRAM
AND PRIMARY
ILLUMINATED SEMI-GRAPHIC DIAGRAM TURBINE & SECONDARY
CONTROL
PANEL
ATORS
ILLUMINATED SEMI-GRAPHIC DIAGRAM GENERATOR & DISTRIBUTION
11FT 0'
START UP & CIRCUIT BREAKER CHECK LIST PANEL
4FT 0'
10FT 0'
2
LIST OF MATERIAL
APPLICATION
H. FRENCH
CONTROL CONSOLE
U. S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.
M 11594-76



CABLE SIZE & CURRENT RATING
INSIDE VAPOR CONTAINER
N.E.C.

PENETRATION No. COND.	CABLE SIZE	AMPS AT 122° F.	AMP LOAD	USE
P1-3/C	4/0	172.5		P.C. PUMP MOTOR
P1-3/C	4/0	172.5	360	P.C. PUMP MOTOR
P3-1/C	4/0			GROUND ROD
P4-24/C	#9 (19x22)	28	23	PRESSURIZER HEATER
P5-40/C	#12 (19x25)	15	-1	ROD DRIVE MOTOR
P6-40/C	#12 (19x25)	15	-1	ROD DRIVE MOTOR
A1-40/C	#12 (19x25)	15	-1	ROD DRIVE INSTRUMENTS
A2-40/C	#12 (19x25)	15	-1	ROD DRIVE INSTRUMENTS
A3-22/C	#9 (19x22)	28	20 MAX.	LIGHTING CIRCUITS
A4-40/C	#16 IRON-CONSTANTAN THERMOCOUPLES			
A5-40/C	#12 (19x25)	15	-1	INSTRUMENTATION
A6-40/C	#12 (19x25)	15	-1	INSTRUMENTATION
N1-9/C	COAXIAL-RG 71/U			ION CHAMBER LEADS
N2-9/C	COAXIAL-RG 71/U			ION CHAMBER LEADS
N3-4/C	COAXIAL-RG 59/U			BF-3 - FISSION ION CHAMBERS

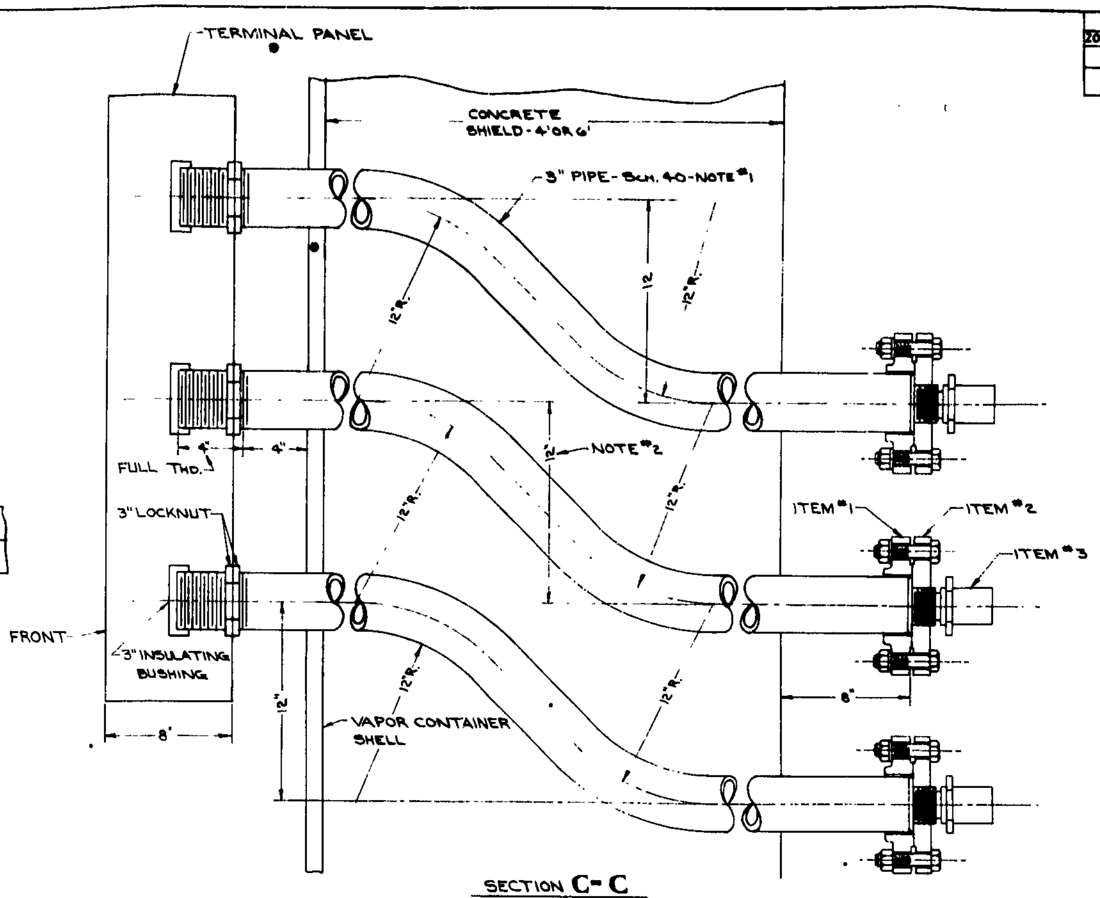
1- ALL PIPE PENETRATIONS ARE 3" SCH. 40 EXCEPT P3 WHICH IS 1 1/2" SCH. 40.

2- PENETRATION PIPES ARE OFF-SET 12" ON THEIR DIAMETERS. OFF-SET TO START AT VAPOR CONTAINER SHELL.

3- CABLES A3 THRU A6 TERMINATE IN TERMINAL BOARD LOCATED ON SOUTH WALL OF V.C.

1

REVISIONS				
ZONE	SYM.	DESCRIPTION	DATE	APPROVAL
	A	10'-0"R. WAS 20'-0"R	5/26/60	
	B	ADDED DRG NO. M11594-74 ECO 0292	3-3-61 ETH	



SECTION C-C

ITEM #1 - 300# SLIP-ON FLANGE
ITEM #2 - 300# BLIND FLANGE-DRILLED & TAPPED WITH LEFT HAND THREADS FOR PYLE-NATIONAL RECEPTACLE
ITEM #3 - PYLE-NATIONAL ENVIRONMENT RESISTANT ELECTRICAL CONNECTORS

REF. M11594-60

ROUND)

R
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OR
RUMENTS
TRUMENTS
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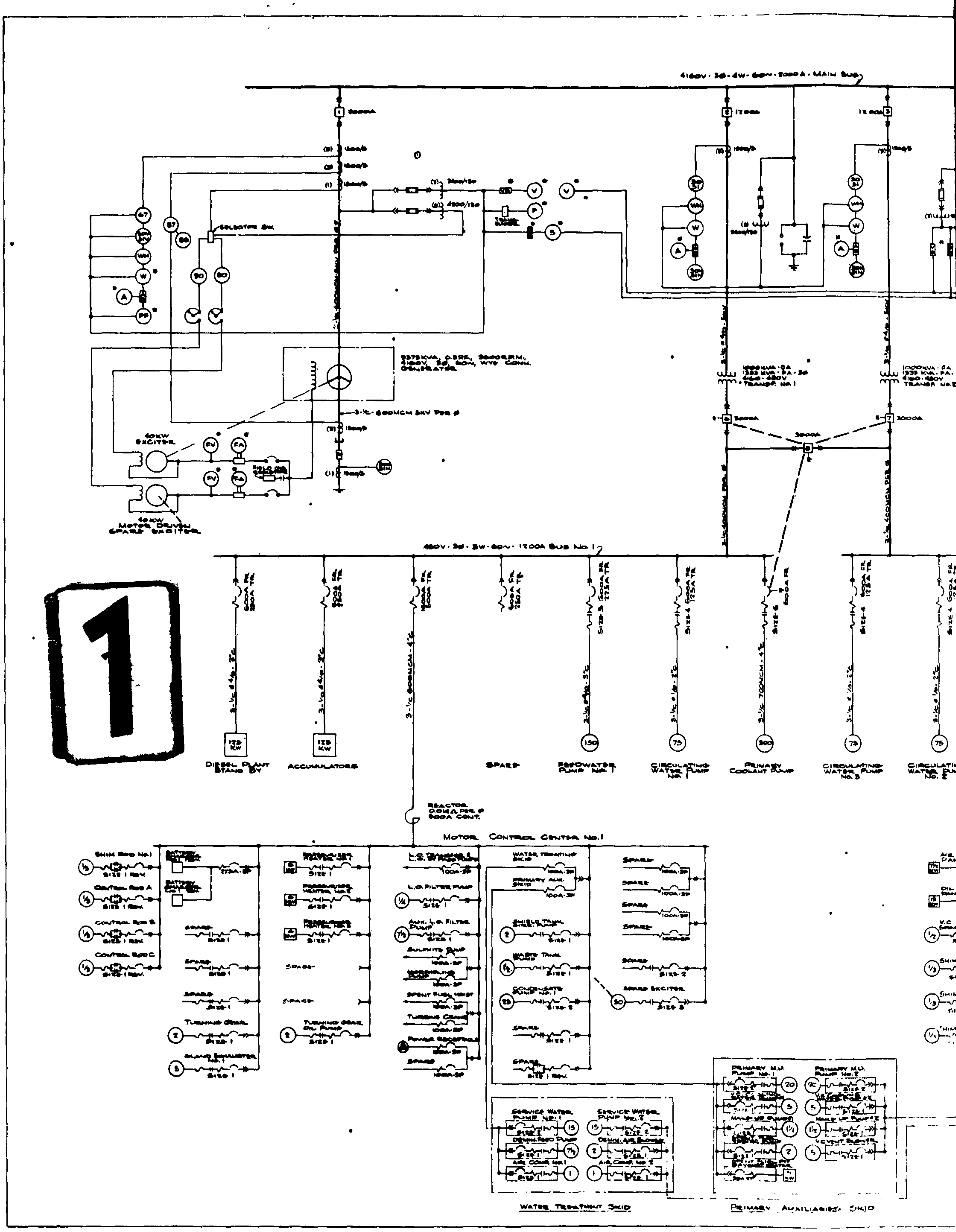
ON
ION
EADS
EADS
BERS

M11594-74 B

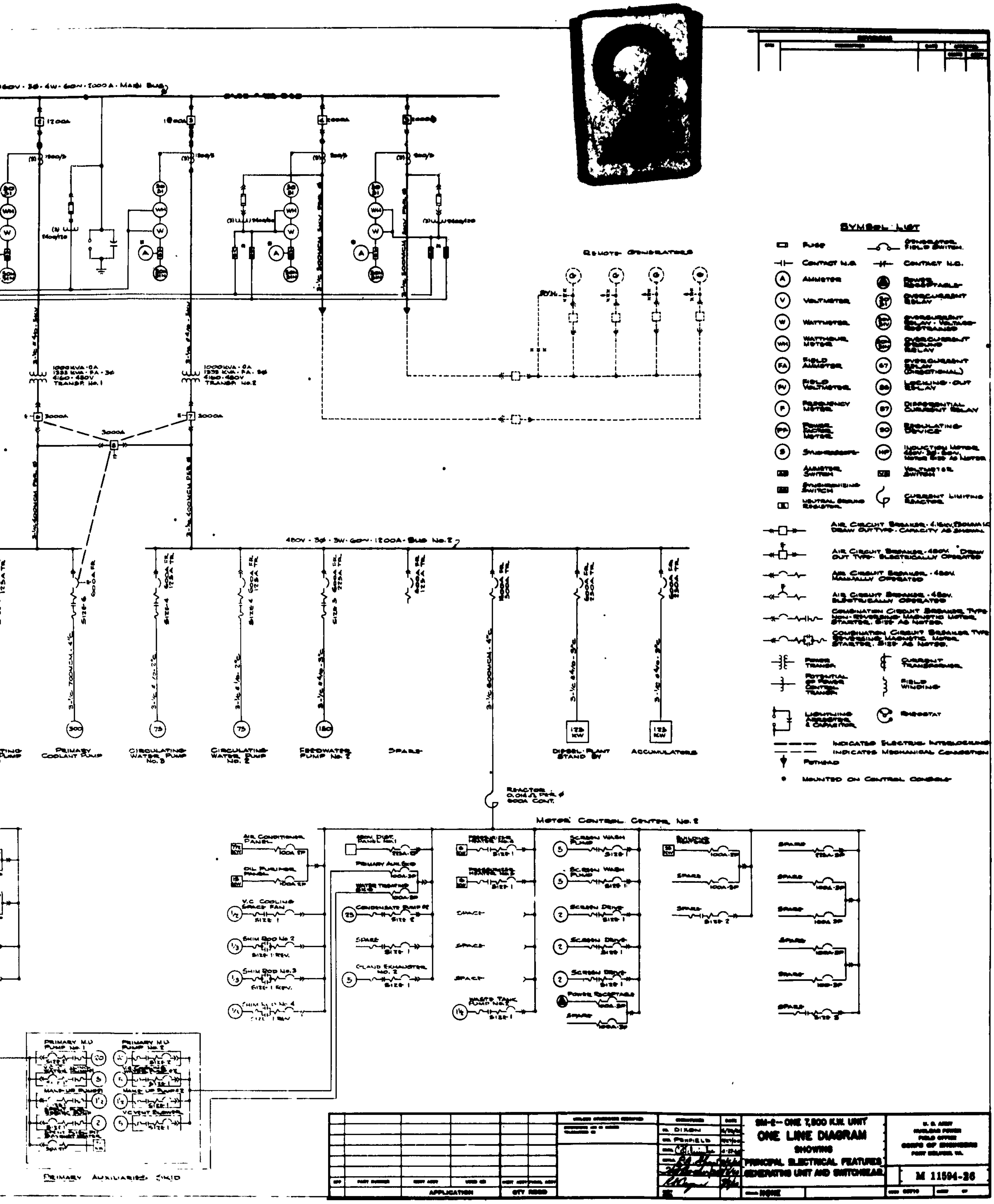
ELECTRICAL PENETRATION
VAPOR CONTAINER
SM-2

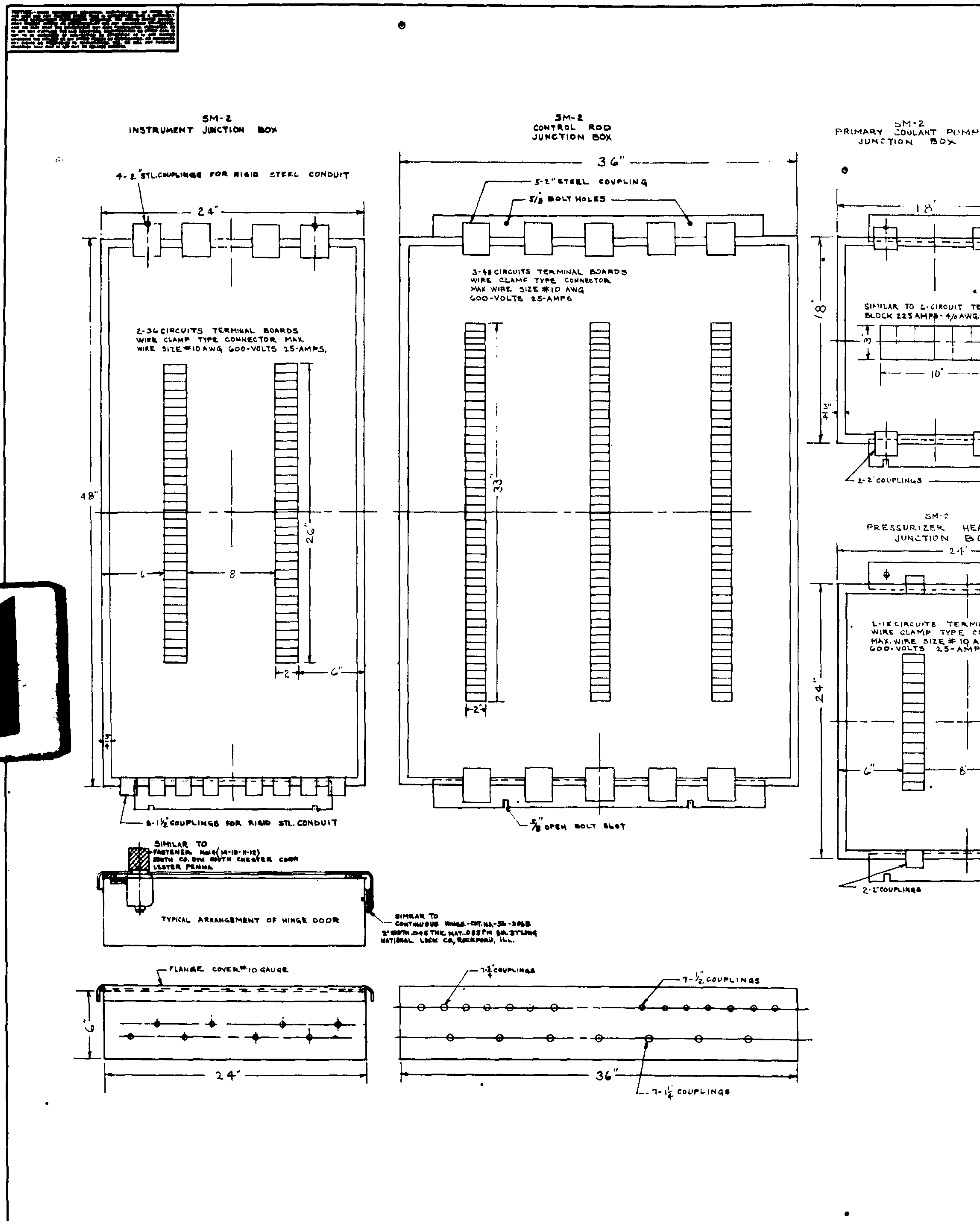
U. S. ARMY
NUCLEAR POWER FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-74

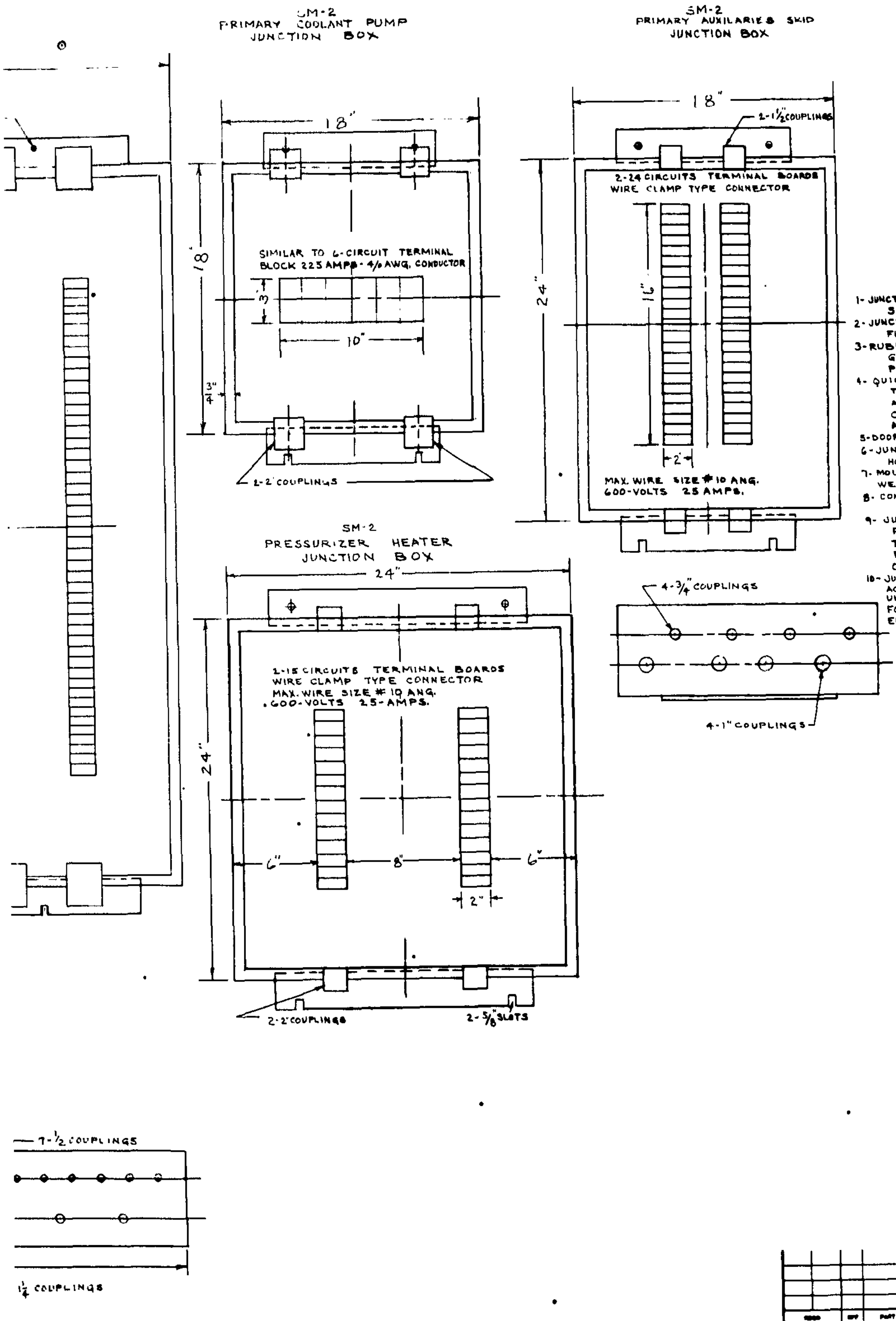


4160V · 3Ø · 4W · 60N · 2000A · MAIN BUS
SELECTOR SW.
40KW EXCITER
40KW MOTOR DRIVEN SPARE EXCITER
480V · 3Ø · 3W · 60N · 1200A BUS No. 1
1
DIESEL PLANT STAND BY
ACCUMULATORS
SPARE
FEEDWATER PUMP No. 1
CIRCULATING WATER PUMP No. 1
PRIMARY COOLANT PUMP
CIRCULATING WATER PUMP No. 3
MOTOR CONTROL CENTER No. 1
SHIM ROD No. 1
CONTROL ROD A
CONTROL ROD B
CONTROL ROD C
TURNING GEAR
TURNING GEAR OIL PUMP
L.O. FILTER PUMP
SPARE EXCITER
SERVICE WATER PUMP No. 1
SERVICE WATER PUMP No. 2
WATER TREATMENT SKID
PRIMARY AUXILIARIES SKID


SYMBOL LIST
Fuse
Ammeter
Voltmeter
Wattmeter
Watthour Meter
Field Ammeter
Field Voltmeter
Frequency Meter
Synchroscope
Ammeter Switch
Synchronizing Switch
Current Limiting Reactor
Power Transf.
Current Transformer
Field Winding
Rheostat
Lightning Arrester & Capacitor
Indicates Electrical Interlocking
Indicates Mechanical Connection
Pothead
Mounted on Control Console
Remote Generators
1000KVA - OA
1333 KVA - FA - 3φ
4160 - 480V
Transf. No. 1
Transf. No. 2
3000A
480V - 3φ - 3W - 60~ - 1200A - Bus No. 2
Primary Coolant Pump
Circulating Water Pump No. 3
Circulating Water Pump No. 2
Feedwater Pump No. 2
Spare
Diesel Plant Stand By
Accumulators
Motor Control Center No. 2
Primary Auxiliaries Skid
Primary M.U. Pump No. 1
Primary M.U. Pump No. 2
SM-2 — ONE 7,500 K.W. UNIT
ONE LINE DIAGRAM
SHOWING
PRINCIPAL ELECTRICAL FEATURES
GENERATING UNIT AND SWITCHGEAR
M 11594-26
NONE
APPLICATION
QTY REQD


SM-2
INSTRUMENT JUNCTION BOX
4-2" STL. COUPLINGS FOR RIGID STEEL CONDUIT
24"
2-36 CIRCUITS TERMINAL BOARDS
WIRE CLAMP TYPE CONNECTOR MAX.
WIRE SIZE #10 AWG 600-VOLTS 25-AMPS.
48"
26"
6
8
2
6"
8-1½" COUPLINGS FOR RIGID STL. CONDUIT
SM-2
CONTROL ROD
JUNCTION BOX
36"
5-2" STEEL COUPLING
5/8 BOLT HOLES
3-48 CIRCUITS TERMINAL BOARDS
WIRE CLAMP TYPE CONNECTOR
MAX WIRE SIZE #10 AWG
600-VOLTS 25-AMPS
33"
2"
5/8" OPEN BOLT SLOT
SM-2
PRIMARY COOLANT PUMP
JUNCTION BOX
18"
18"
3"
10"
2-2" COUPLINGS
SM-2
PRESSURIZER HEA
JUNCTION BO
24"
24"
6"
8"
2-2" COUPLINGS
SIMILAR TO
FASTENER
LESTER PENNA
TYPICAL ARRANGEMENT OF HINGE DOOR
SIMILAR TO
NATIONAL LOCK CO., ROCKFORD, ILL.
FLANGE COVER #10 GAUGE
6"
24"
7-¾" COUPLINGS
7-½" COUPLINGS
36"
7-1¼" COUPLINGS

1



JUNCTION BOX REQUIREMENTS

1- JUNCTION BOXES TO BE FABRICATED FROM SHEET STEEL.
2- JUNCTION BOXES TO BE FLANGED TYPE WITH FLANGED TYPE DOORS.
3- RUBBER TYPE (NEOPHRENE OR POLYETHYLENE) GASKET TO SEAL DOOR TO PROVIDE WATER PROOF ; ENCLOSURE
4- QUICK CLOSING TYPE DOOR FASTENERS TO BE USED, FASTENERS TO BE LOCATED AT DOOR CORNERS, FASTENERS BETWEEN CORNERS TO BE NOT MORE THAN 24 IN. APART.
5- DOOR HINGES TO BE LOCATED ON RIGHT SIDE
6- JUNCTION BOXES TO BE GALVANIZED BY THE HOT DIP PROCESSES.
7- MOUNTING BRACKETS TO BE CONTINUOUS WELDED TO BACK SURFACE OF BOXES
8- CONDUIT COUPLINGS TO BE CONTINUOUS WELDED
9- JUNCTION BOXES TO BE FABRICATED FROM # 10 GAUGE (.120 IN.) SHEET STEEL TO PROVIDE NECESSARY STRENGTH FOR PLUG IN RECEPTACLES LOCATED ON TOP OF JUNCTION BOXES.
10- JUNCTION BOXES TO BE FABRICATED ACCORDING TO THE STANDARDS OF THE UNDERWRITERS LABORATORIES, INC. FOR USE IN ACCORDANCE WITH THE NATIONAL ELECTRIC CODE

2

LIST OF MATERIAL

APPLICATION | QTY REQD

H. Fromm 9-1-60
R.S.Berg 9-1-60
9-1-60

ELECTRICAL WIRING
(SM-2 PRIMARY SKID JUNCTION BOXES)

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F M 11594-75

11.0 FUEL HANDLING TOOLS

The objective of the design program for the SM-2 fuel handling tools is to develop tools which are functional, easily handled, simple to operate and low in cost. Tools are to be designed for handling the core components during the re-fueling procedure.

The most practical approach is to design lightweight, portable, manually-operated tools handling as many functions as possible. Since the re-fueling procedure is carried out remotely under water, a long handle will be common to all tools. Interchangeable end tools fitting the same handle would reduce the cost and simplify the storage of the tools.

Tools used for repetitious operations such as removing fuel elements, absorbers, and control rod caps could utilize a handle which the operator could telescope during operation. The tool could be crane-supported and operated at platform level. No counter-balance weights would be required. The operator would have precise control of the tool and efficiency would be greater than with a fixed-length handle.

A design was developed for telescoping two tubes by means of a crank, cable, and pulley system. This design, shown on M11594-84, utilizes a crank, rod and tube to actuate a screw which operates the jaws of the interchangeable tools. These tools are attached to the end of the handle with a knurled lock nut.

The 1/8 in. diam stainless steel cable is in two sections with one end of each section fastened to a fixed lifting pin at the top end of the inside tube. The other end of one cable is fastened to a take-up drum in the control box and the other end of the second cable is passed over the fixed pulley at the bottom of the outside tube and back to the second take-up drum. Thus, for extending the inside tube, the cable around the fixed pulley pulls the tube down when the take-up drum is rotated and the second take-up drum lets out the cable which is fastened to the lifting pin. The reverse of this operation retracts the inside tube.

A ratchet on one cable take-up drum holds the lower telescoping tube at any extended length and a safety lock prevents the ratchet from being unlocked accidentally while the tool is being handled.

The latch operating rod slides inside a square tube when the handle sections are telescoped. A square bar fastended to the bottom end of the rod inside the square tube transmits the torque to rotate the screw on the tool.

Two tube supports with guide bushings are provided to prevent bowing of the latch rod and tube.

Two Teflon guide rings are provided for the telescoping tube sections, one at the top end of the inside tube and one at the bottom end of the outside tube.

This telescoping handle can be used with three tools, the combination fuel handling tool, the control rod cap tool, and the retrieving tool.

The combination fuel handling tool, M11594-82, is capable of handling the stationary fuel elements, control rod fuel elements, and absorbers. It also can be used to replace the photo-neutron startup source in the core.

The combination fuel handling tool consists of a pair of interlocking jaws, connected by a toggle linkage to a nut and screw. The latch rod in the telescoping handle is used to rotate the screw which pulls up the nut and opens the jaws of the tool. The fuel elements have a lifting tab with a hole to be gripped with the jaws of the tool. The absorber has a straight lifting pin which is gripped with the tool. The photo-neutron source is supplied with a wire loop to be gripped with the tool.

The retrieving tool, M11594-83, is provided for picking up objects which have fallen into the reactor. It consists of a pair of claw-type jaws actuated in the same manner as the jaws of the combination fuel handling tool. It is capable of picking up a dropped fuel handling element. If necessary, side pieces can be attached to the jaws to form a bucket for picking up small objects.

The control rod cap tool, M11594-80, is used with the telescoping handle to remove the control rod caps from the top end of the control rod tubes and transport them to the dummy tubes for storage.

The cap tool consists of a spring-loaded cap body and a pair of vertical latches. The tool is placed over the cap and depressed about 1/2 in. to disengage a locking pin from a notched slot in the cap sleeve. Then the cap is rotated 45° counterclockwise to disengage the lugs which lock the cap to the control rod tube. At the same time, the latches slide over the cam surface of the cap and catch under the extended edge of the cap. Release of pressure on the cap allows the spring-loaded cap body and pin to engage the notch in the slot and lock the cap on the tool. The cap can then be lifted free of the tube.

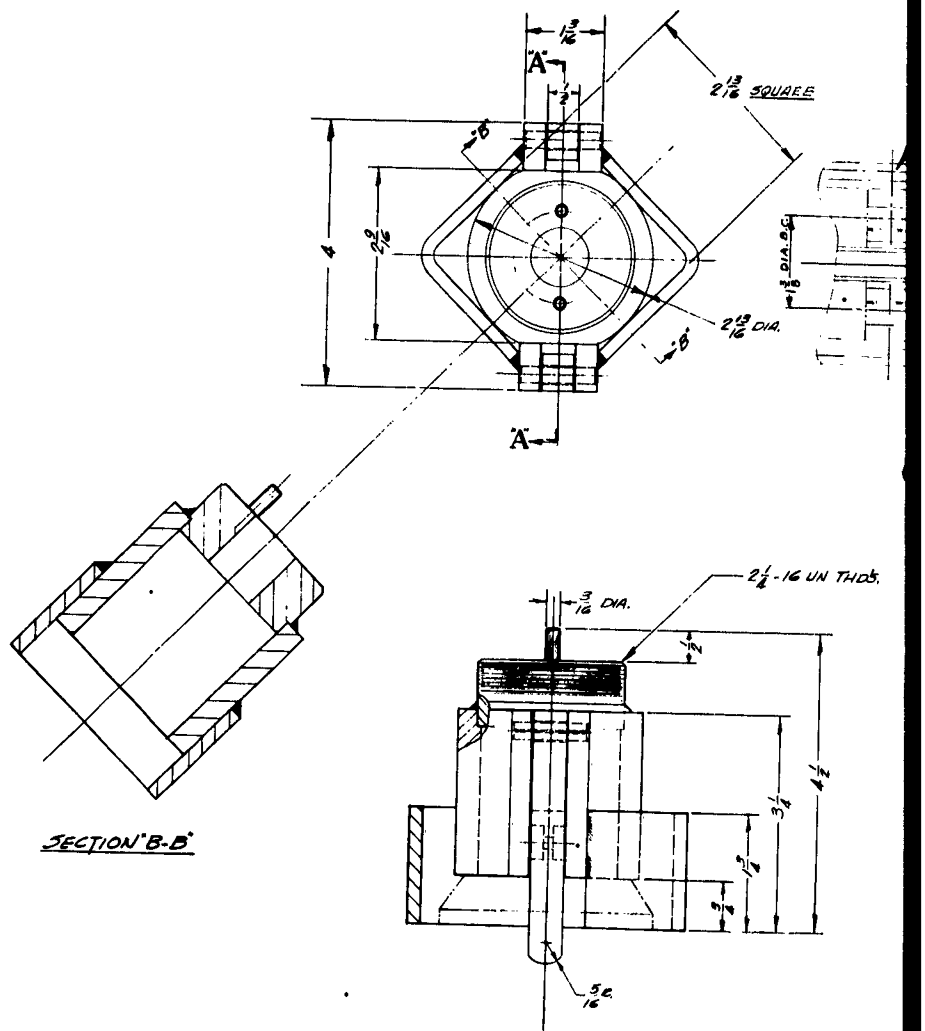
A tool is required for lifting the core grid plate which weighs approximately 300 pounds. The telescoping handle is not adequate to lift this plate so another handle, M11594-85 was designed to be used for this purpose.

This handle consists of a tube with a knurled lock nut on the bottom end for attachment to the tool. A slot in the end engages a pin on the tool to prevent relative rotation and possible unthreading of the lock nut.

To remove the core grid plate, the grid plate lifting tool, M11594-81, is placed on top of the grid plate with the four radial arms of the tool positioned adjacent to the latches at the corners of the grid plate. The latches, operated with the grid plate latch tool, are rotated clockwise to release them from the tie rods and to position them over the radial lifting arms. The grid plate is then free and can be lifted from the tie rods.

The grid plate latch tool, M-11594-86, is used with handle M-11594-85. This tool has a socket end which fits over the hexagonal nut welded on the latch. Simultaneously, a pin on the tool depresses a spring-loaded plunger out of a hole in the latch and the tool and latch can be rotated clockwise. A second hole in the latch receives the plunger when the tool is removed and the latch is locked in position on the lifting tool.

[redacted]

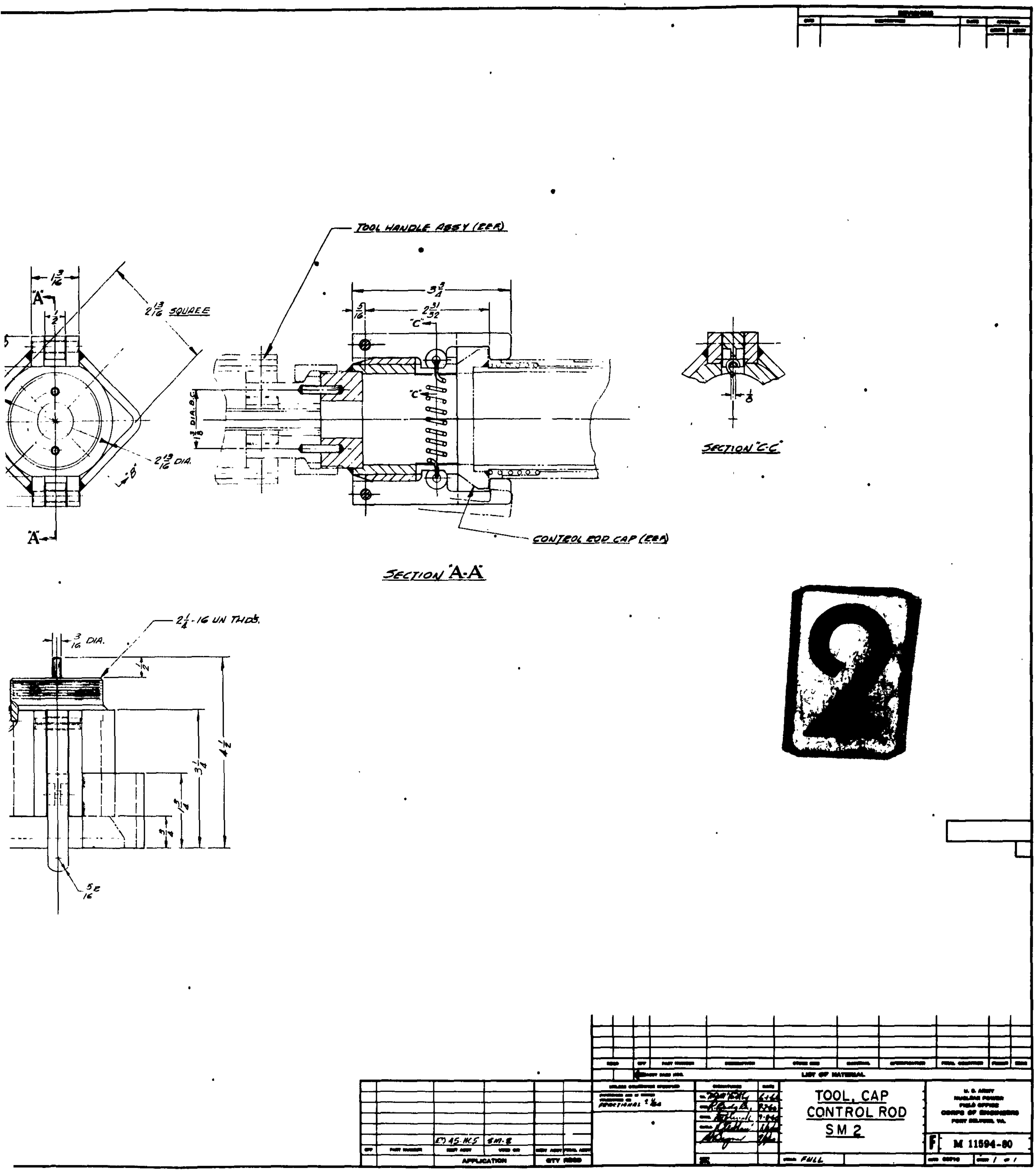


1

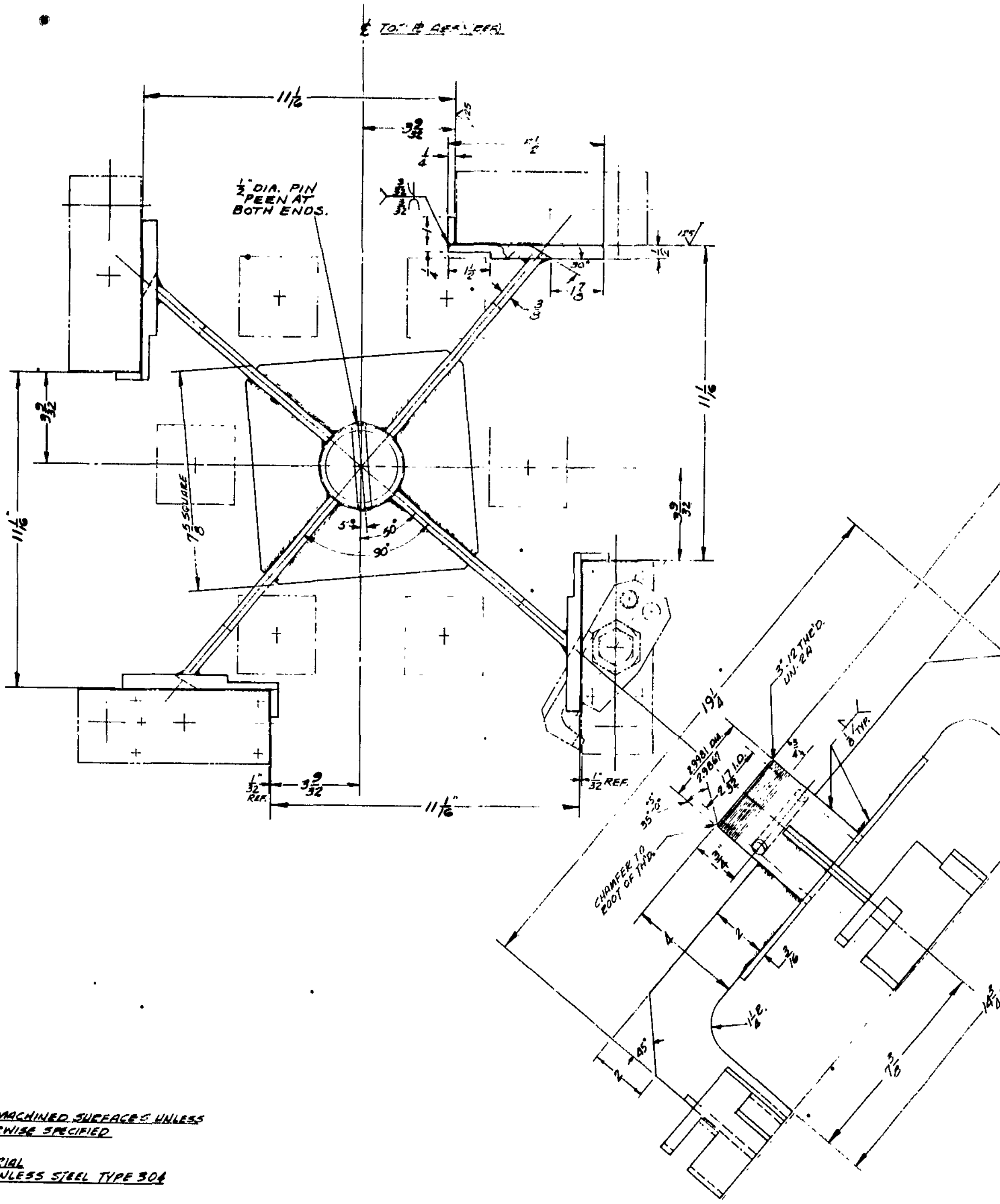
250 √ ALL MACHINED SURFACES UNLESS OTHERWISE SPECIFIED

MATERIAL
STAINLESS STEEL TYPE 304

ALL WELDS $\frac{1}{8}$" FILLET, UNLESS OTHERWISE SPECIFIED.


TOOL HANDLE ASSY (REF)
CONTROL ROD CAP (REF)
SECTION "A-A"
SECTION "C-C"
2 13/16 SQUARE
2 13/16 DIA.
2 1/4 -16 UN THDS.
3/16 DIA.
3 3/4
2 31/32
4 1/2
3 1/4
LIST OF MATERIAL
FRACTIONAL ± 1/64
ED 45-MC5
SM-2
APPLICATION
QTY REQD
TOOL, CAP
CONTROL ROD
SM 2
U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.
F
M 11594-80
FULL
1 of 1

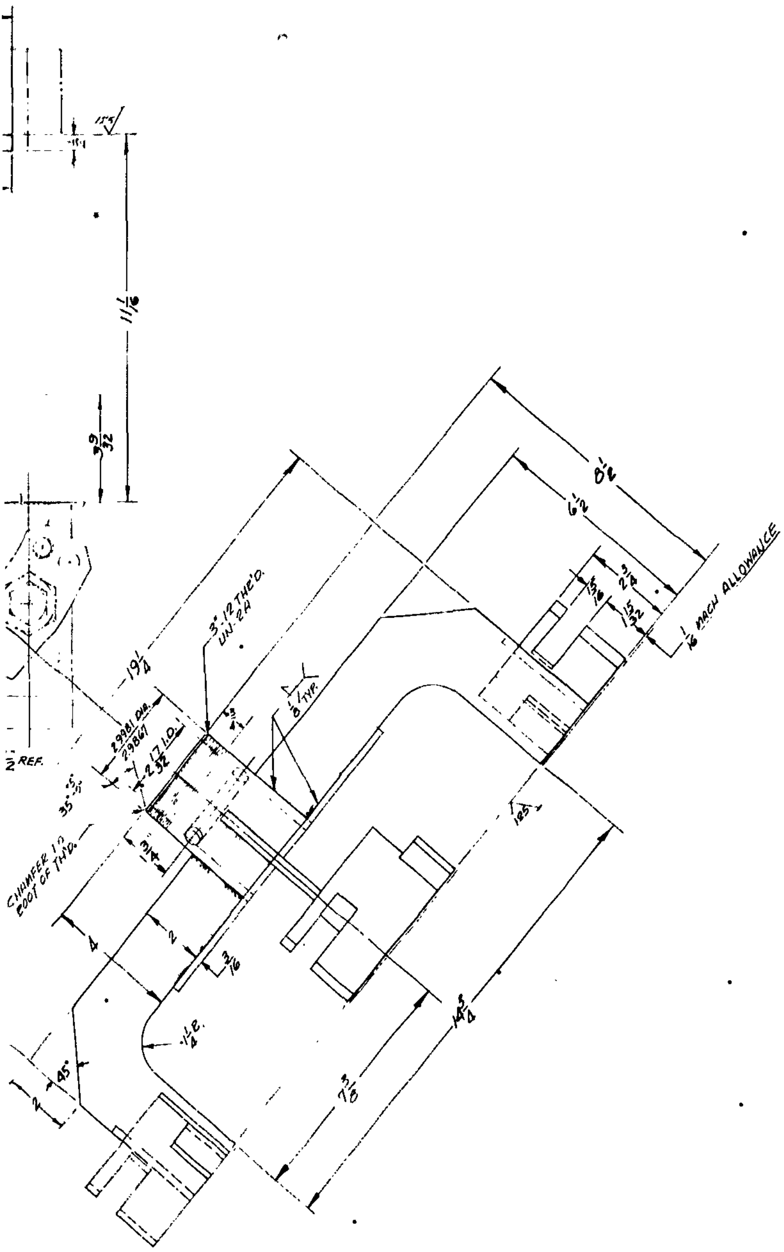
[redacted]



1

250 ALL MACHINED SURFACES UNLESS OTHERWISE SPECIFIED

MATERIAL
STAINLESS STEEL TYPE 304



LIST OF MATERIAL

TOOL,
GRID PLATE LIFTING

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

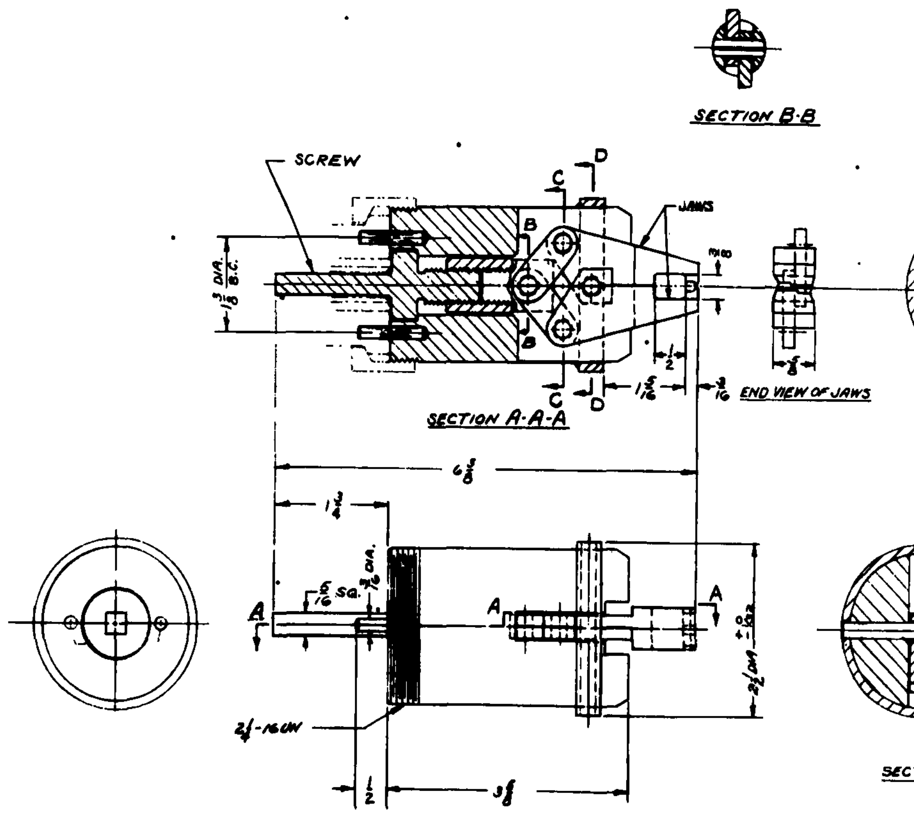
F M 11594-61

SM-2

APPLICATION

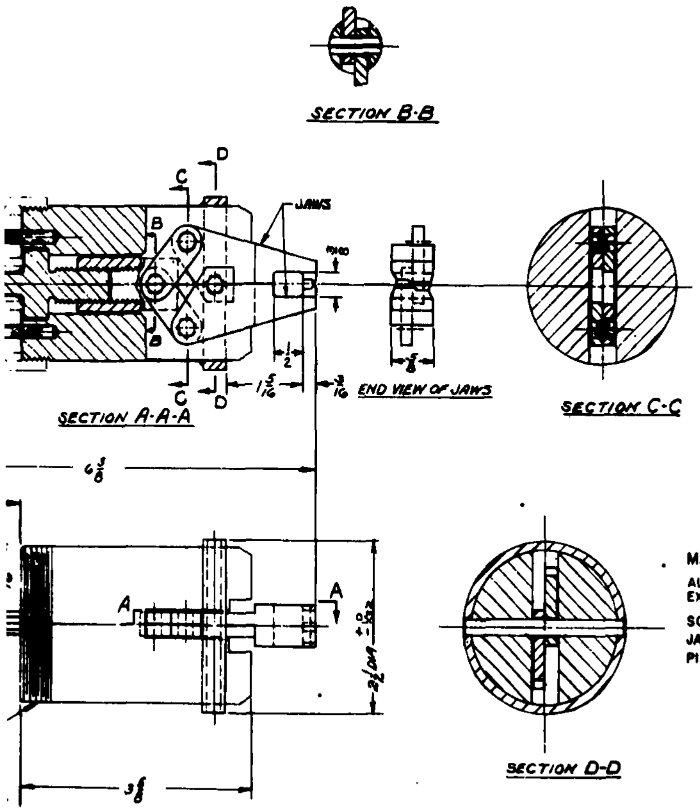
QTY REQD

6":1'



SECTION B-B
SCREW
JAWS
SECTION A-A-A
END VIEW OF JAWS
A
B
C
D
$1\frac{3}{8}$ DIA. B.C.
$1\frac{5}{16}$
$6\frac{3}{8}$
$1\frac{3}{4}$
$\frac{5}{16}$ SQ.
$2\frac{1}{4}$-16UN
$\frac{1}{2}$

1



MATERIAL:

ALL MATERIAL TO BE 6061-T6 ALUMINUM EXCEPT AS NOTED FOR THE FOLLOWING PARTS:

SCREW - TYPE 304 STN. STL.
JAWS - TYPE 304 STN. STL.
PINS - TYPE 302 OR 303 STN. STL.

LIST OF MATERIAL

APPLICATION | QTY REQD

H. FRENCH

TOOL, COMBINATION FUEL HANDLING

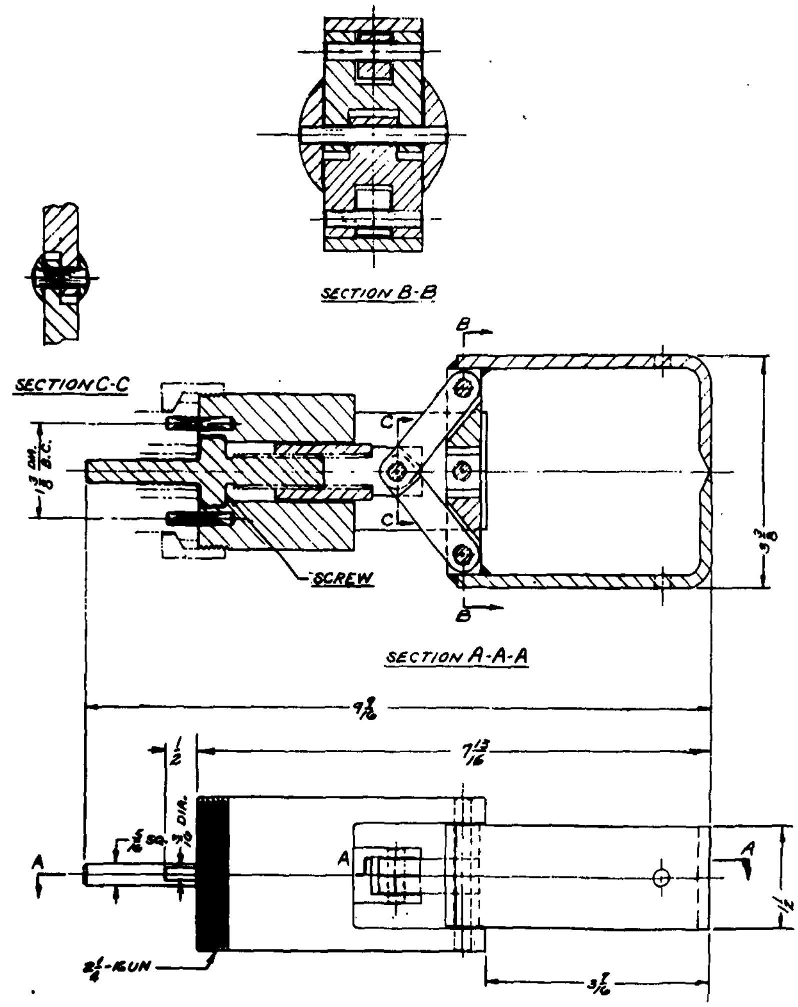
U. S. ARMY
NUCLEAR POWER FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F | M 11594-82

SCALE FULL

2

[redacted]



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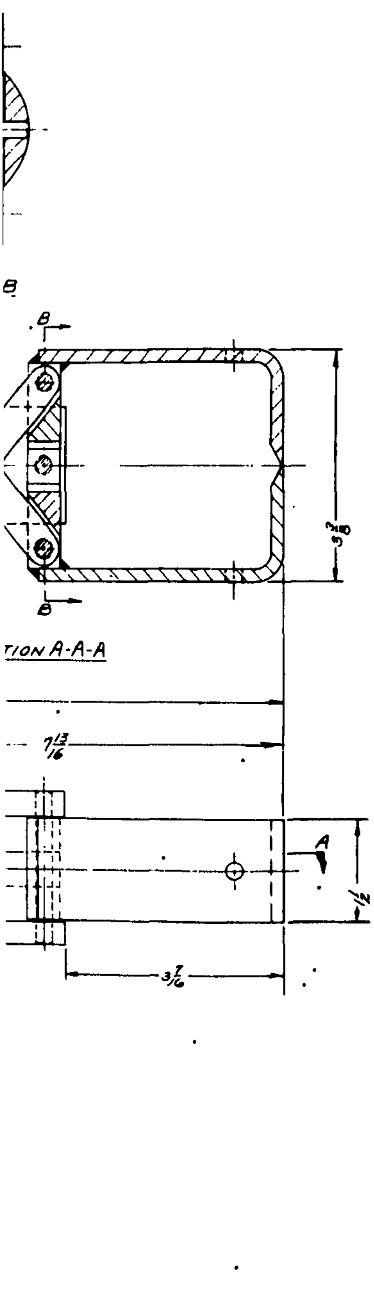


MATERIAL:
ALL MATERIAL TO BE 6061-T6 ALUMINUM EXCEPT
AS NOTED FOR THE FOLLOWING PARTS:
SCREW – TYPE 304 STN. STL.
PINS – TYPE 302 OR 303 STN. STL.

LIST OF MATERIAL

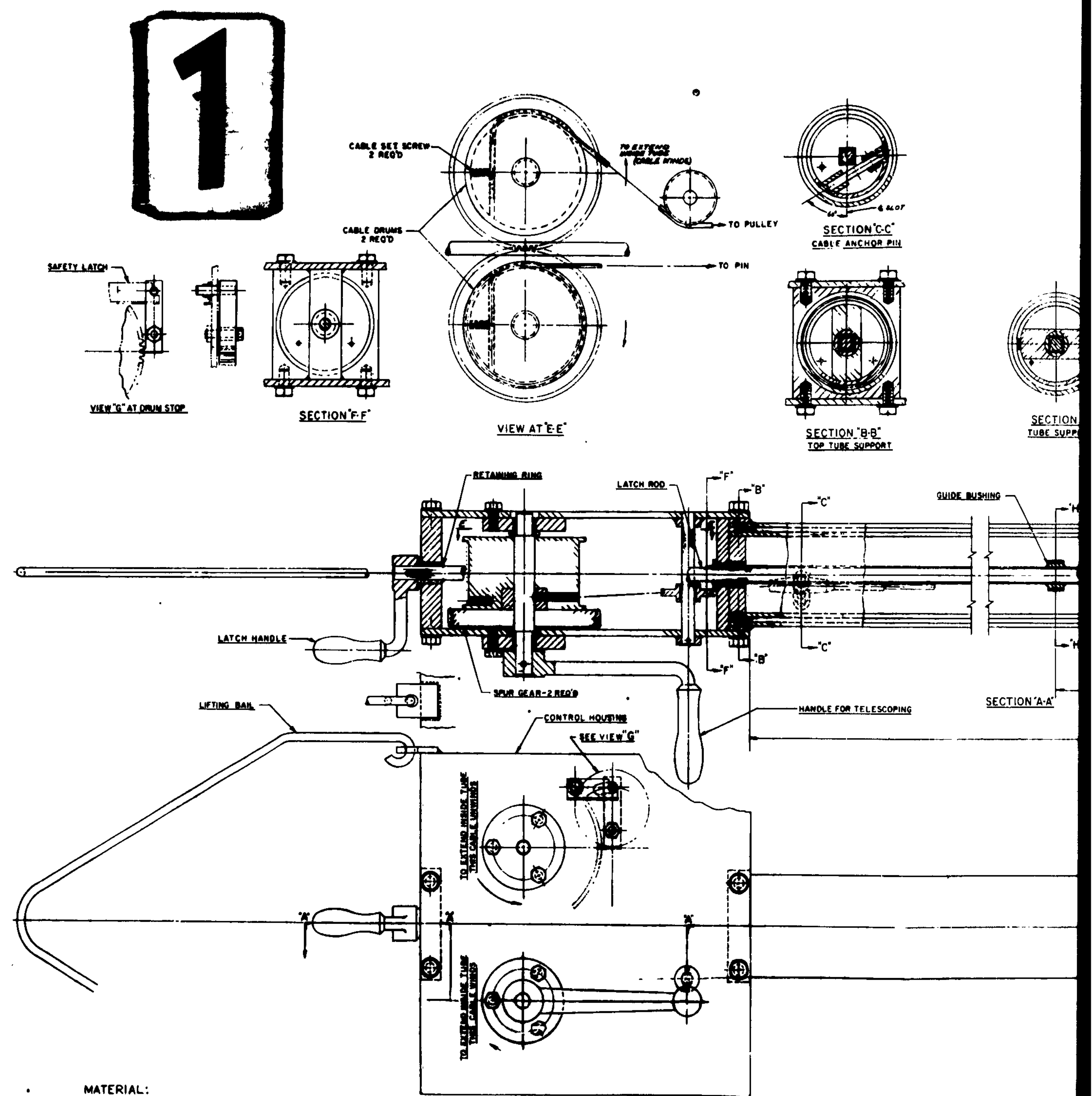
APPLICATION | QTY REQD

TOOL, RETRIEVING

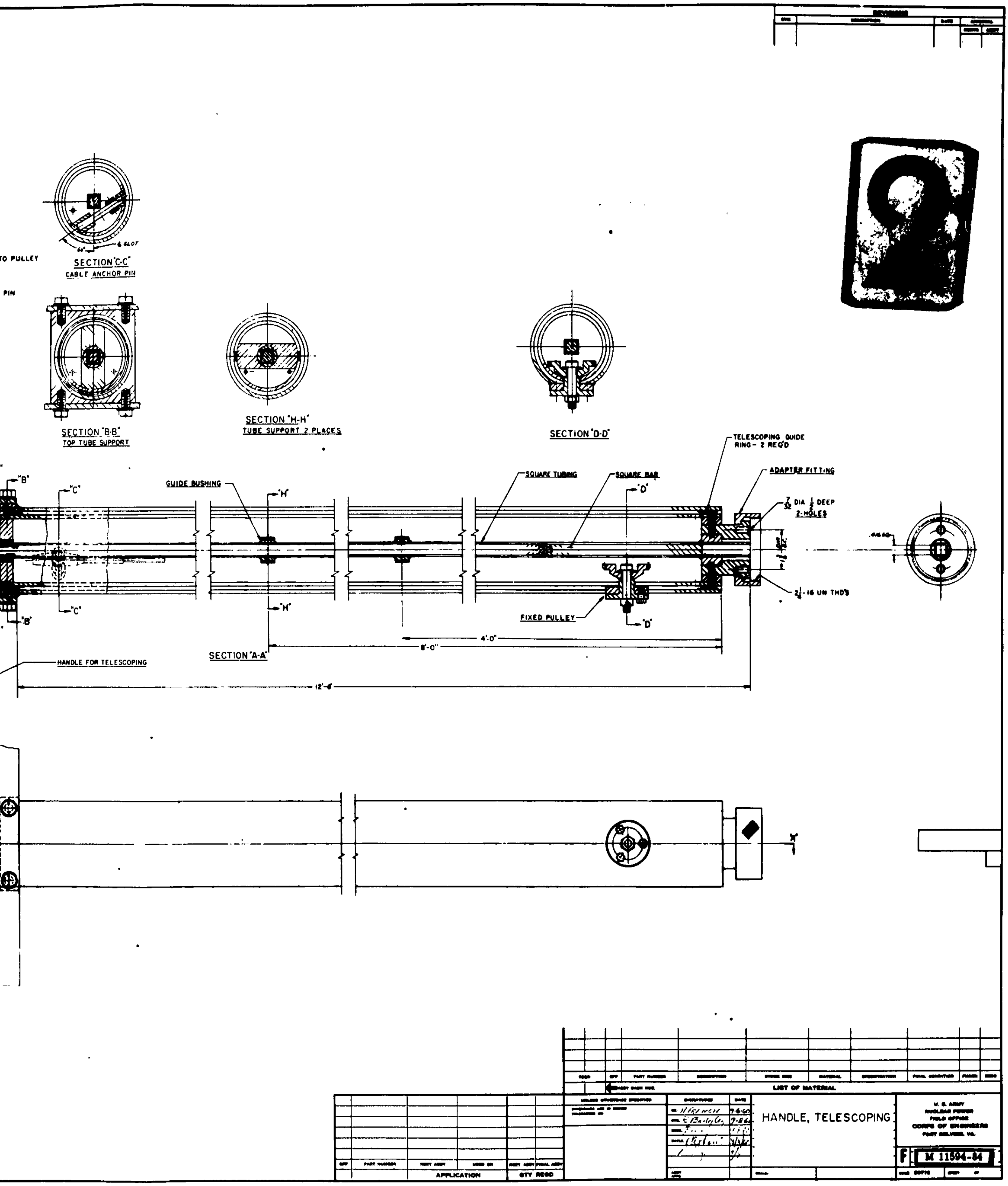
U.S. ARMY
CORPS OF ENGINEERS

F M 11504-03

FULL



MATERIAL:
ALL MATERIAL TO BE 6061-T6 ALUMINUM EXCEPT
AS NOTED FOR THE FOLLOWING PARTS:
LATCH ROD AND SQUARE BAR - TYPE 304 STN. STL.
SQUARE TUBING - TYPE 304 STN. STL.
ADAPTER FITTING - TYPE 304 STN STL.
GEARS - CAST IRON OR 1020 CARB STL., PARKERIZED.
CRANK HANDLES - MALL. IRON.
SHAFTS - TYPE 302 STN. STL.
BUSHINGS - BRONZE.
TELESCOPING GUIDE RINGS - TEFLON
CAPSCREWS, NUTS LOCKWASHERS, PINS - TYPE 302 OR 303 STN. STL.
CABLE - TYPE 302 OR 303 STN. STL.
LIFTING BAIL - 1020 CARBON STEEL, PARKERIZED.


TO PULLEY
PIN
SECTION 'C-C'
CABLE ANCHOR PIN
SECTION 'B-B'
TOP TUBE SUPPORT
SECTION 'H-H'
TUBE SUPPORT 2 PLACES
SECTION 'D-D'
TELESCOPING GUIDE RING - 2 REQ'D
ADAPTER FITTING
SQUARE TUBING
SQUARE BAR
GUIDE BUSHING
7/32 DIA 1/2 DEEP 2-HOLES
2 1/4-16 UN THD'S
FIXED PULLEY
4'-0"
8'-0"
SECTION 'A-A'
HANDLE FOR TELESCOPING
12'-6"
LIST OF MATERIAL
APPLICATION
QTY REQD
HANDLE, TELESCOPING
U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.
F
M 11594-84

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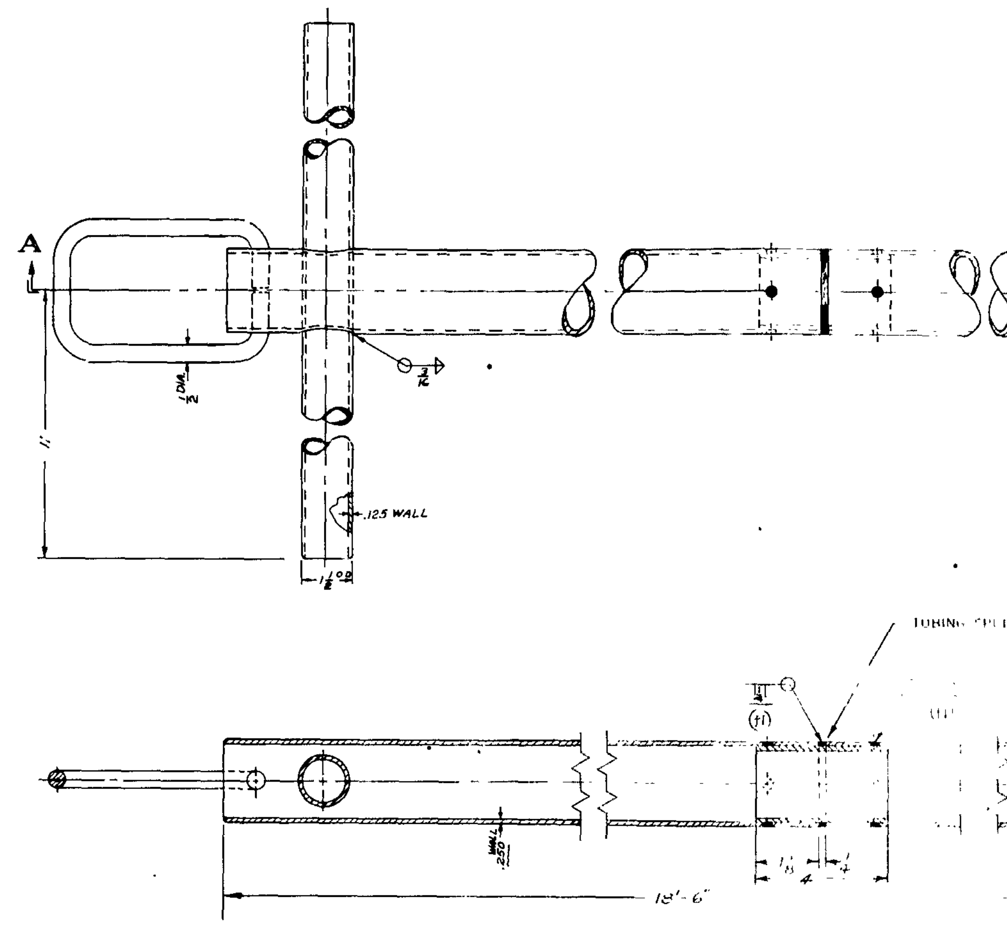


SECTION A-A

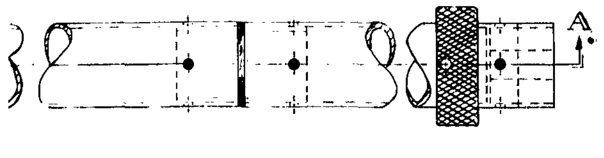
1

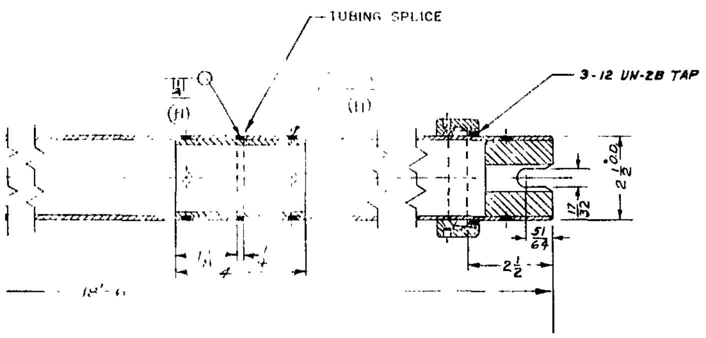
MATERIAL.

ALL MATERIAL TO BE 6061-T6 ALU
AS NOTED FOR THE FOLLOWING P

CONNECTING NUT- TYPE 304 STN
LIFTING EYE - CARBON STEEL SA

REVISIONS





TION A-A

MATERIAL:

ALL MATERIAL TO BE 6061-T6 ALUMINUM EXCEPT
AS NOTED FOR THE FOLLOWING PARTS:

CONNECTING NUT- TYPE 304 STN. STL.
LIFTING EYE - CARBON STEEL SAE-1020, PARKERIZED.

LIST OF MATERIAL

APPLICATION | QTY REQD

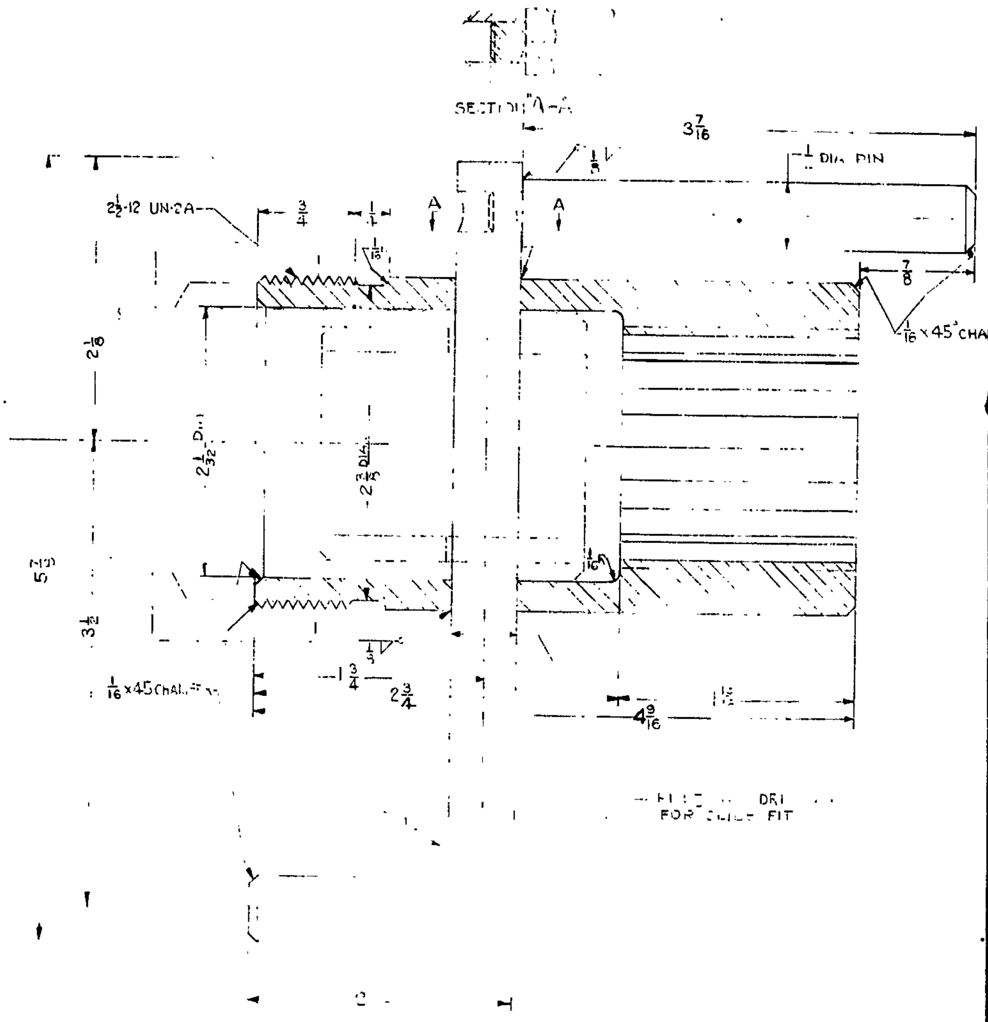
SM-2

HANDLE, TOOL

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F M 11594-85

[redacted]



1

NOTES

√ALL OVER

PARKERIZE

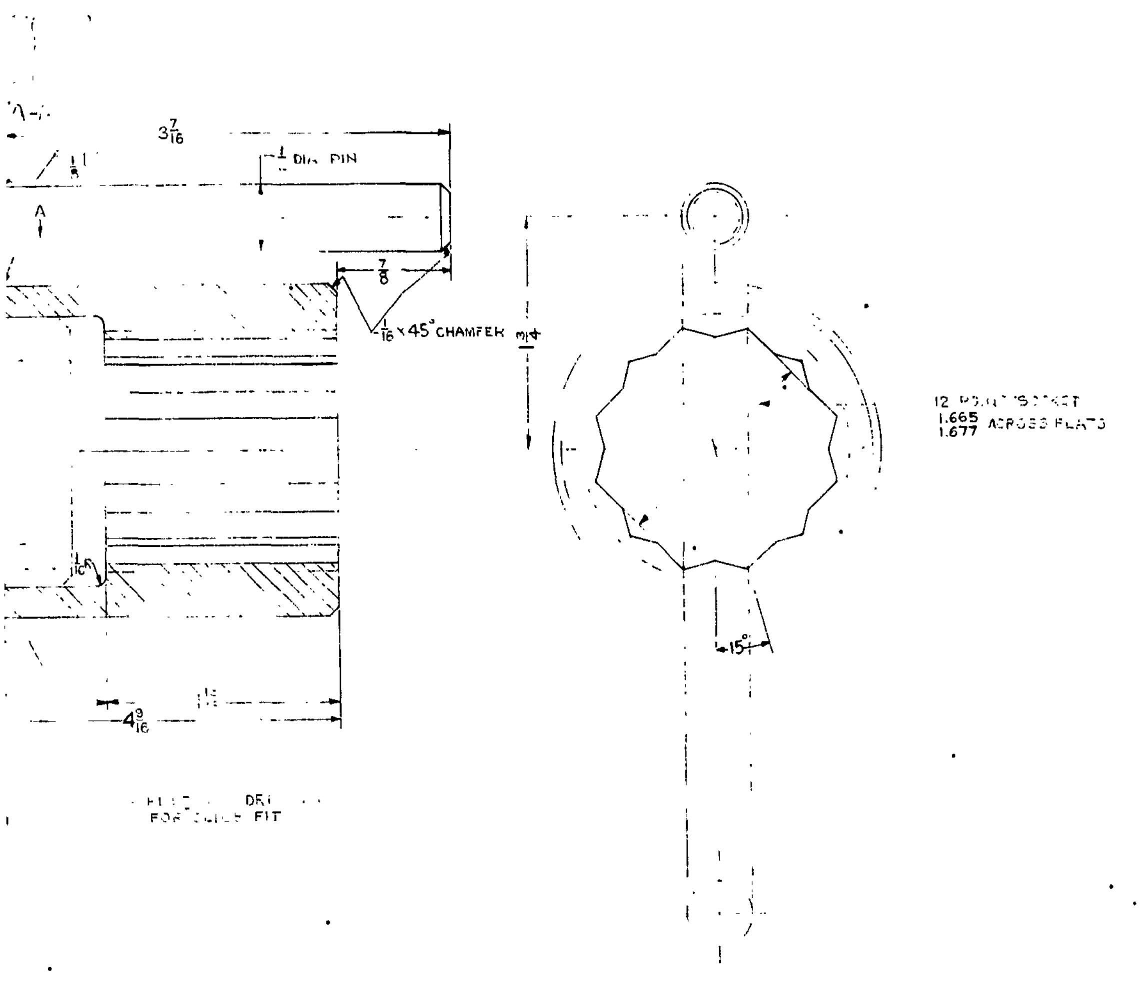
MATL

STEEL SAE 41

1575°F-1½ H WATER QUEN

29/35 R

SYM	DESCRIPTION	DATE	APPROVAL
A	ADDED DWG NO M11594-86 ECO 0292	ETN	



STEEL SAE 41
29/35

M11594-86

A

		SM-2		
QTY	PART NUMBER	NEXT ASSY	USED ON	QTY REQD
		APPLICATION		

LIST OF MATERIAL

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

86

2 x SIZE

12.0 WASTE DISPOSAL SYSTEM

12.1 LIQUID WASTE DISPOSAL

An SM-2 waste disposal system must be capable of processing at a reasonable cost both high and low solids waste containing varying amounts of radioactivity. The system must also be provided with some flexibility as the SM-2 site has not been firmly established. Since even an unlimited supply of water will not permit the established activity tolerances to be achieved by dilution, some processing is required. Evaporation, with its high decontamination efficiency, appears to be most suitable for processing the range of liquid wastes that will be encountered in the operation and decontamination of the SM-2 plant. A detailed discussion concerning the factors which lead to the choice of evaporation may be found in APAE No. 56.(1)

Hot liquid wastes will be derived from the following sources:

1. Laboratory, sampling and laundry.

2. Decontamination solutions and rinses.

3. Primary coolant contaminated by a failed fuel element.

Processing of liquid wastes will follow the procedure described below:

1. All wastes will first drain into a 2500 gal. tank.

2. A sump pump will pump the liquid waste from the 2500 gal tank into one of two 5000 gal storage tanks. The pH of the liquid collected in the 2500 gal tank will be checked before it is pumped to the larger tanks. If it is acidic, caustic will be added to raise the pH.

3. During normal operation, about 2500 gal will be allowed to accumulate in one of the 5000 gal tanks. Additional waste will then be diverted to the other 5000 gal tank while the waste in the first tank is allowed to decay for 1 to 2 months.

4. After the liquid waste has undergone a period of undisturbed decay, it will be pumped to a holding tank and subsequently to a stainless steel evaporator.

5. The concentrated bottoms accumulating in the evaporator will be drawn off to a drumming station to be packaged in prefabricated steel drums shielded by concrete (on all sides except the top). The drums will contain vermiculite. After being filled, the drums will be sealed with ready-mix concrete and stored at the site until they are shipped to a

waste processing center for final disposal.

6. The condensate from the evaporator will be collected in a tank. The condensate will be pumped from the collection tank into the main condenser cooling water discharge line at a rate which will ensure that the water being discharged into the lake or river will have an activity of less than $1 \times 10^{-7} \mu c/cc$.

7. If a laboratory analysis of samples from the waste collection tanks shows that the activity of the liquids is below $1 \times 10^{-7} \mu c/cc$, the evaporator can be by-passed and the liquid sent directly to the main condenser discharge line.

8. Decontamination cycles 1-3 which includes both decontaminating solutions and one water rinse, will be collected in one of the 5000 gal tanks. A steam heating coil in the tank will allow the temperature to be raised to 85°C so that the reaction between the decontaminating solutions, which yields a slurry of manganese dioxide, can be completed within a few hours.

9. The manganese dioxide slurry will be mixed by a mechanical agitator while it is being pumped to the holding tank. A mechanical agitator in the holding tank will keep the slurry well mixed as it is being fed to the evaporator.

10. Decontamination cycles 4-6 which are water rinses will be discharged directly into the main condenser cooling water line if the activity can be diluted to below $1 \times 10^{-7} \mu c/cc$. If these activity levels cannot be obtained the rinses will be processed in the evaporator.

11. Connections have been provided for a demineralizer. Therefore, in the event the evaporator becomes inoperable, waste processing could be done by demineralization. Also, if the activity of the liquid waste is low but not low enough to be diluted sufficiently, it could be cleaned up in the demineralizer instead of being processed by the evaporator.

During normal plant operations, liquid wastes will accumulate at the rate of 470 gal/mo if the sample stream from the continuous oxygen analyzer is put back into the primary system. Otherwise, the accumulation of liquid wastes may range up to 1610 gal per month. Close to 10000 gal of liquid will be collected during decontamination. The evaporator with its 50 gph capacity should be adequate.

12.2 GASEOUS WASTE DISPOSAL

All tanks holding radioactive liquid waste, as well as the liquid waste evaporator, will be vented to a compressor intake manifold. When a small positive pressure is built up in the vent lines, the compressor will be actuated. The gases will be drawn through an absolute filter and compressed in cylinders capable of containing up to 1000 psi. The gases stored in the cylinders will undergo a 6 month period of undisturbed decay. At the end of this period, Kr85 will be the only radioactive gas present in any significant concentration. Under these conditions, it will be permissible to release the gases to the atmosphere at 3×10^{-7} μc/cc.

REFERENCES

1. Zegger, J. L., "Waste Disposal Considerations for the SM-2 Design", APAE No. 56, May 13, 1960.

13.0 SPENT FUEL SYSTEM

13.1 TRANSFER SYSTEM

The spent fuel transfer system (Dwg. M11594-59) incorporates the advantages of a horizontal transfer tube. The spent fuel chute under the 45° angle as used in the SM-1 and SM-1A vapor container designs requires that the bottom of the pit be located considerably below the elevation required for safe storage.

The distance between the reactor and spent fuel tank center lines is 24 ft 9 in. Transfer of the fuel elements is by means of a sled (Dwg. M11594-61) actuated through a 6 in. stainless steel Schedule 40 pipe by a steel band, used similar to a plumber's or electrician's snake. Integrity of spent fuel tank and vapor container is maintained by a 6 in., 150 lb stainless steel gate valve at each end of the pipe. Valve stem extensions are provided so that these valves can be opened or closed from the operating platforms in the vapor container and over spent fuel tank. Sled is so designed that it will pass through the valves when they are in the fully open position.

Since compacted backfill will serve as secondary shielding between the vapor container and spent fuel tank, a 15" corrugated culvert pipe will enclose the 6 in. transfer pipe to prevent any differential settlement. It will be necessary to install a clamp joint in the middle of this 15 in. pipe.

The fuel element carriage is so designed that it will remain in a horizontal position during the transfer, but at either end, a lightweight basket within the carriage can be tilted into a vertical position and held there by a gravity-acting stop so that elements can be inserted or removed. The basket can be returned to the horizontal position by a push of the stop with a tool.

Personnel will perform spent fuel transfer from platforms in the vapor container and over the spent fuel tank. No special tools beyond the regular element handlers are contemplated.

13.2 SPENT FUEL PIT

The spent fuel pit (Dwg. M11594-59) is a shop fabricated vessel of stainless steel. It is cylindrical in shape with inside diameter of 108 in. and depth of 16 ft, 6 in. A determination of earth pressures and fill surcharges will determine the thickness and bracing required, and it now seems evident that these values will probably dictate that the pit be designed as the inside form for a concrete structure to carry these loads.

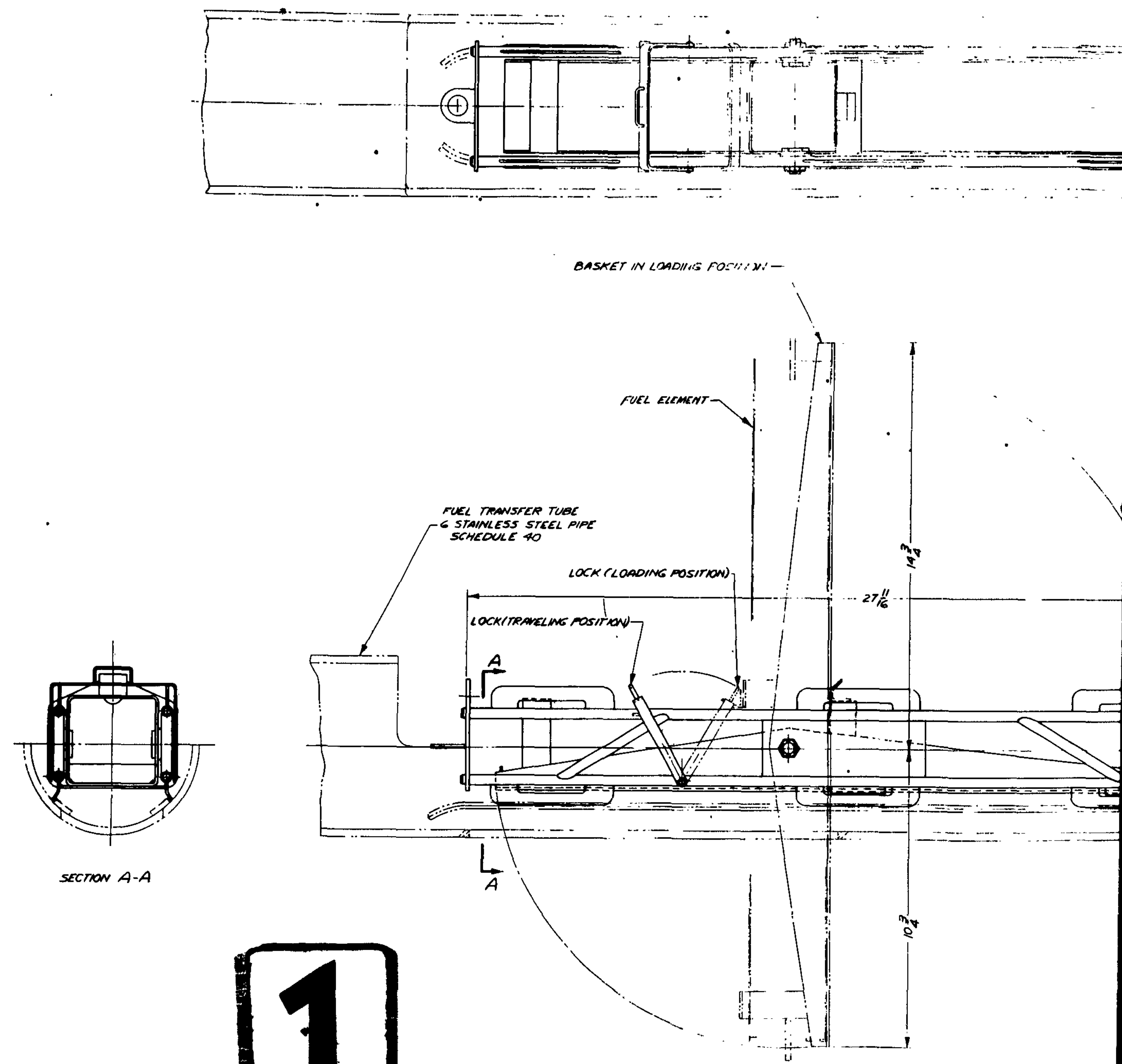
The spent fuel pit is as close as possible to the vapor container consistent with secondary shielding requirements.

Design of the spent fuel tank includes receptacles for 90 elements plus space for a spent fuel shipping cask.

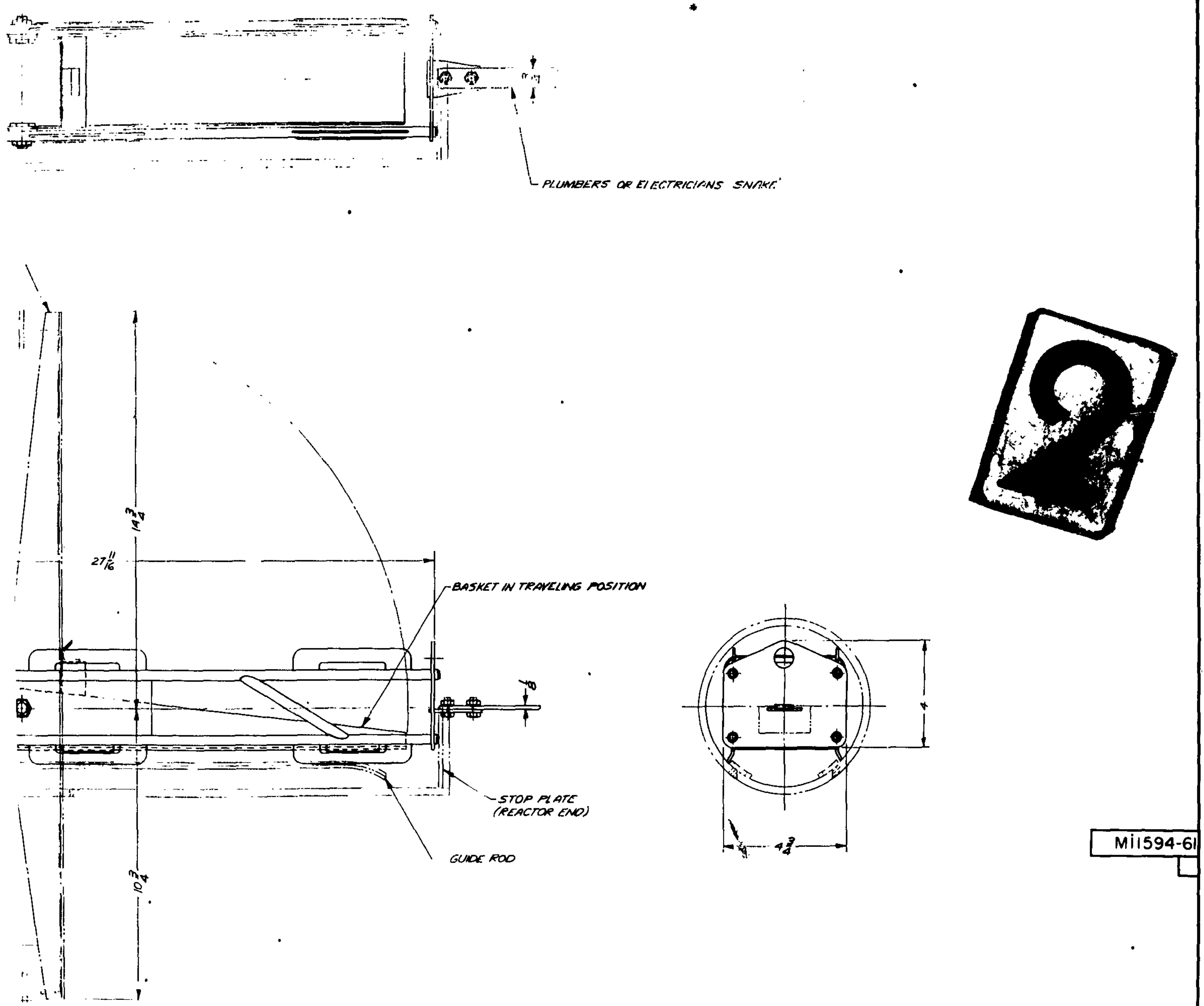
The tank cover will consist of removable sections which will facilitate spent fuel transfer. An overhead crane is provided over the pit to aid during transfer operations.

Spent fuel tank water temperature will be regulated by pumping water through a cooler when temperature reaches 150^{o}F or over bayonet heaters to prevent freezing. Pump is designed for 10 gpm at 43 psig. Tank will be equipped with a filtering system to maintain purity of water. No demineralizer is intended to be used, but provision is made to include a unit in the system.

[redacted]


BASKET IN LOADING POSITION
FUEL ELEMENT
FUEL TRANSFER TUBE
6 STAINLESS STEEL PIPE
SCHEDULE 40
LOCK (LOADING POSITION)
LOCK (TRAVELING POSITION)
27 11/16
14 3/4
10 3/4
A
A
SECTION A-A

1



LIST OF MATERIAL

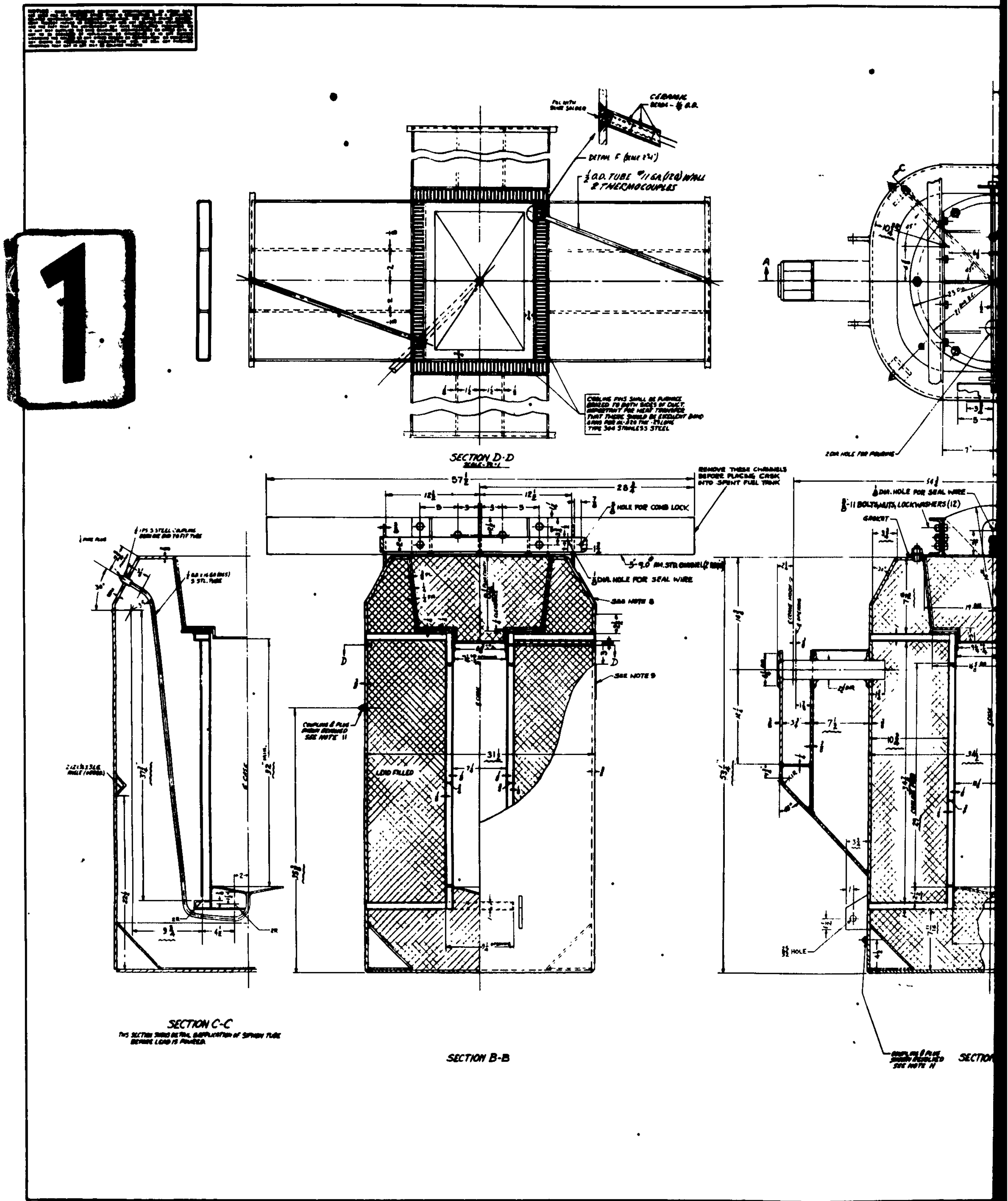
APPLICATION

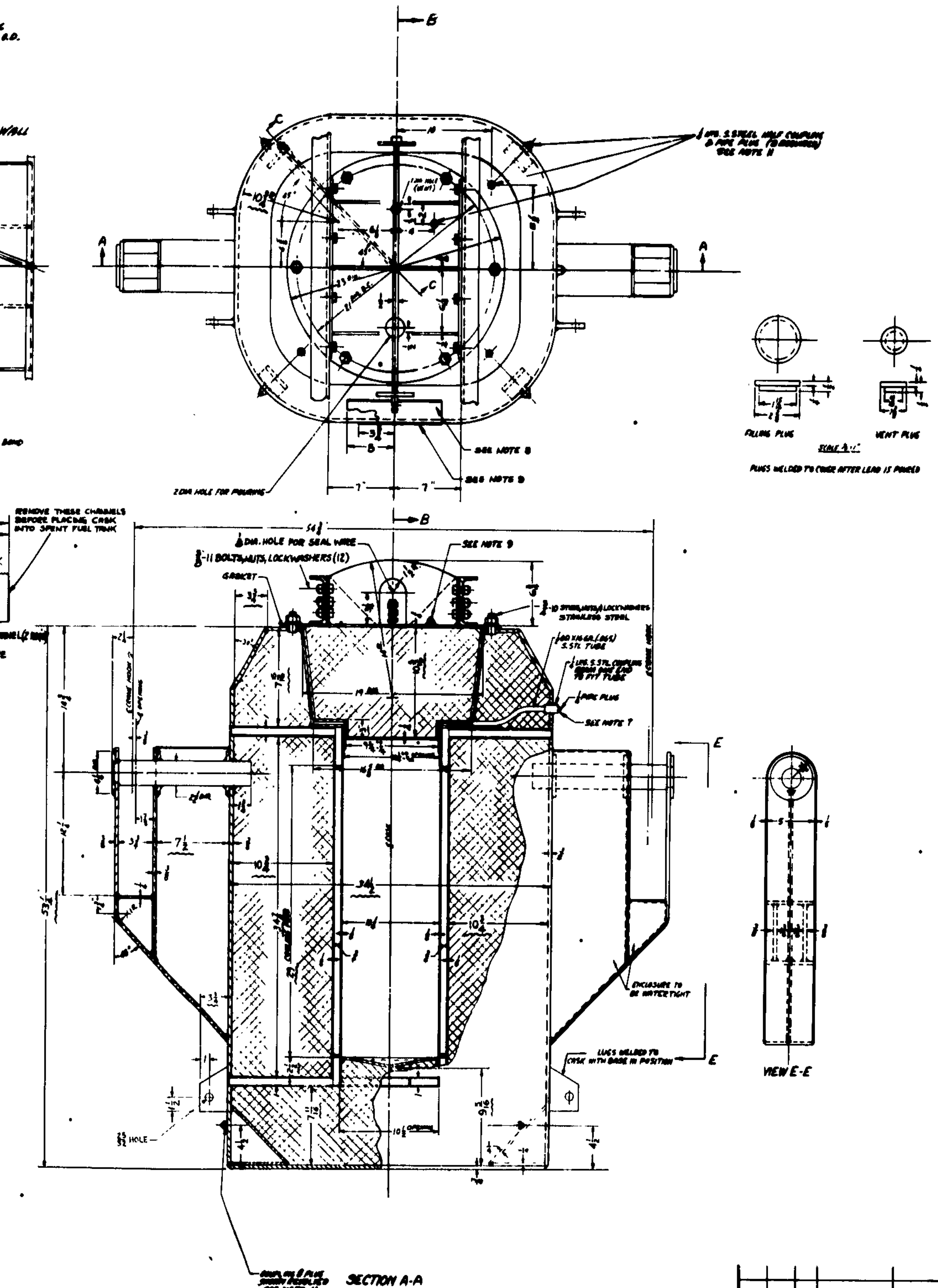
QTY REQD

SPENT FUEL
SLED & BASKET

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-61


1
CERAMIC
DETAIL F
2 THERMOCOUPLES
SECTION D-D
REMOVE THESE CHANNELS BEFORE PLACING CASK INTO SPENT FUEL TANK
HOLE FOR COMB LOCK
DIA. HOLE FOR SEAL WIRE
SEE NOTE 8
SEE NOTE 9
LEAD FILLED
COUPLING & PLUG SEE NOTE 11
DIA. HOLE FOR SEAL WIRE
GASKET
SECTION C-C
SECTION B-B
SECTION



NOTES

1- COMPLETE CASK ASSY & PLUG LIFTING CHANNELS ARE TO BE 304 STAINLESS STEEL
2- INSIDE OF PLUG HOLE & OUTSIDE OF PLUG TO BE FINISHED TO GIVE ... ROUNDNESS & CLOSE FIT ... PLATES & GUSSETS TO BE WELDED TO PLUG BEFORE MACHINING - ALL SHARP EDGES TO BE ROUNDED OFF.
3- ALL WELDS TO BE GROUND & POLISHED SMOOTH
4- ALL WELDS TO BE WATERTIGHT
5- BOTTOM PLATE TO BE WELDED ON AFTER LEAD IS POURED.
6- COMPRESSED AIR CONNECTION FOR DRAINING CAVITY (MAX. 5 PSI AIR)
7- LOCATION FOR NAME PLATE - SEE DWG. A9-48-2048
8- LOCATION FOR CASK WEIGHTS PLATE - SEE DWG. A9-48-2038
9- HYDROSTATIC TEST - TEST CAVITY TO 50 PSIG THROUGH SYPHON DRAIN OR COMPRESSED AIR FITTING
10- WELD LEAK TEST - TEST WELDS TO 50 PSIG HELIUM THROUGH NOTED FITTINGS & MONITOR WELDS WITH HELIUM LEAK DETECTOR
11- ALL WELDING WILL CONFORM TO ASME CODE VIII STAMPING IS NOT REQUIRED.
12- ALL STAINLESS STEEL NUTS & BOLTS WILL BE FLASH HARD CHROME PLATED ABOUT .0001 INCH THICK.

LIST OF MATERIAL

APPLICATION

SHIPPING CASK
SPENT FUEL

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

F M 11594-95

SCALE 2" = 1'-0"

14.0 PRIMARY SHIELDING DESIGN

14.0 LIST OF TABLES

14.0 PRIMARY SHIELDING DESIGN

14.1 GENERAL

AP Note 167[1] presents the initial analysis of shielding for the SM-2 reactor vessel using steel rings, water, concrete, lead shot and lead brick as shielding materials. The steel ring-water shielding was eliminated from further consideration because of excessive cost as reported in the cost evaluation presented in APAE Memo 224.[2]

A water annulus for neutron shielding with a surrounding wall of concrete, lead or lead shot for gamma shielding was selected as the best approach to shield the vessel. Accordingly, a design survey was started comparing these materials in a range of thicknesses with the objective of determining an optimum combination of shielding with respect to the space required. The results of this survey are reported in APAE Memo 247.[3]

Concrete shielding is most economical and will be used where space permits, prinicpally in the area between the reactor and the vapor container at the rear of the skid. Because of close clearances between the reactor vessel and the steam generator and primary pump, lead brick shielding will be used in this area. The shielding survey indicates that lead brick shielding requires a minimum allowable water gap of 23 inches. Required lead brick thickness for this water annulus is 7-1/2 inches. Concrete 32-1/2 in. thick with a density of 150 lb/ft^3 is required at the sides of the skid with the water annulus of 23 inches.

The lead and concrete shielding heights are equal. The lead shielding thickness decreases in steps to 7, 5, 3, 2 and 1 in. as shown on Dwg. M11594-90. The inlet and outlet elbow piping penetrations and the control rod drive penetrations are shielded by lead shot contained in boxes designed to provide the required shielding thicknesses.

Shielding during the refueling procedure is provided by an upper shield tank above the reactor containing water to a depth of approximately 17-1/2 ft to the centerline of the core.

The primary shielding structure consists of the lower shield tank, upper shield tank, lead brick shielding, lead shot containment boxes, and the concrete shielding. The upper and lower shield tanks will be shop fabricated. The lower shield tank will be shipped assembled on the primary skid while the upper shield tank will be shipped separately and will be field welded to the top of the lower shield tank. The lead shot containment box for the outlet elbow will be shipped independently as it would project beyond the allowable width of the primary skid if it was shop assembled. Due to shipping weight limitations, the lead brick shielding and lead shot shielding will be shipped independently.

The lower shield tank consists of the inner tank shell, outer tank shell, base plate, expansion joint, cooling coils, dry cap drain header, jackscrews, gussets, and lead shot containment boxes. The inner tank shell also serves as the vessel support ring. It will be necessary to build the lower shield tank in place around the reactor vessel.

Four support lugs on the vessel project through openings in the support shell. On each support lug, a wedge is inserted in the clearance between the top of the lug and the top of the opening. This wedge keeps the vessel held down during shipment, but allows radial expansion of the vessel during startup. A jackscrew is mounted on the support shell at each lug for centering the vessel in the support shell and to keep the vessel retained during shipment. At the site, these jackscrews are removed and replaced with pipe plugs to seal the jackscrew holes against water leakage.

The expansion joint closes off the annular area between the reactor vessel and inner tank shell. It retains the air in the dry cap around the vessel cover and it also retains the upper shield tank water when the dry cap cover is removed during the refueling procedure,

A drain header with four equally spaced outlets adjacent to the expansion joint at the lowest level in the dry cap allows the dry cap to be drained or filled with water.

Cooling coils of one inch schedule 40 pipe are provided around the top of the lower shield tank to remove heat from the shield tank water. The water will be maintained at an approximate temperature of 150 degrees.

Lead shot containers are provided around the inlet elbow, outlet elbow, and control rod drive shaft tubes for shielding these openings in the lead brick.

The upper shield tank consists of the tank shell, base rings, dry cap shell, dry cap cover, cooling coils, lower tank fill tube, and storage racks for the vessel cover, nuts, core grid plate, and control rod caps.

The tank shell retains the shielding water over the reactor during the refueling procedure. The fuel transfer tube to the spent fuel pit is mounted on the tank shell in the field. The hole in the tank shell must be located and cut at assembly. The tank shell has a protruding segment to provide room for storing the vessel cover in a vertical position.

The base ring is the bottom of the upper tank and also the top of the lower tank when the upper shield tank is welded in position. The vessel cover storage rack is mounted on the base ring. Also mounted on the base ring are the dummy studs for storing the seven control rod caps and the twenty-four vessel cover nuts. Twelve studs are provided and two nuts are threaded on each stud.

The dry cap shell and the dry cap cover are provided to retain an insulating blanket of air around the reactor vessel flange and cover. The shell and dry cap cover are insulated to prevent heat loss to the shielding water. The dry cap is designed to reduce the temperature differential through the metal thickness of the vessel cover and flange and thus keep thermal stresses to a minimum.

For refueling the reactor, the dry cap cover must be removed from the upper shield tank and stored in a rack on the vapor container wall. A plate with tapped holes is welded on the top of the dry cap cover to store the capscrews when they are removed from the dry cap flange.

A vent line is connected to the dry cap shell through the insulation to admit air when draining the dry cap.

The dry cap insulation jacket is a transition piece from the lower tank to the upper tank and must be field installed with the insulation.

The core grid plate storage shelf is mounted on the inside of the tank shell. The shelf is hinged to allow tool clearances for removing the dry cap cover capscrews. The self hinges upward and a latch holds the shelf in a vertical position.

Cooling coils of one-inch schedule 40 pipe are mounted at the top of the upper shield tank to remove heat from the shield tank water. The coils are designed to keep the water at a maximum temperature of 150^{o}.

The fill tube for the lower tank is mounted in the upper shield tank. It extends from the base ring of the upper tank to the top of the upper tank. The top end of the tube is flanged for mounting the water level control. A vent line and a water supply line are connected to the top end of the tube.

Seven instrument tube wells are provided for two compensated ion chambers, two BF_3 counters and three safety channels. These wells are made of four inch tubing with 11 gauge wall. A junction box is mounted at the top of each instrument tube well. These wells are installed in the field.

The bottom end of the tube wells for the BF_3 counters and the compensated ion chambers are shielded from excessive radiation by solid lead contained in cylinders. These cylinders are supported in the water annulus by brackets mounted on the vessel support shell. The BF_3 counter has 1-7/8 in. of shielding between it and the reactor and the compensated ion chamber has 5-1/8 in. of shielding.

14.2 SHIELDING DESIGN ANALYSIS

14.2.1 Introduction

An analysis of the SM-2 shielding was performed using the present design which is based upon application of the SM-2 to a nuclear power complex. The high quality power requirement of the complex predicates uncompromised use of the facilities after a maximum credible accident. The analysis is illustrated with parametric data which will permit some variations in the shielding for other applications with little additional effort.

14.2.2 Description of SM-2 Plant

The shielding of the SM-2 reactor system was designed with the proposed application in mind of implementing a system which would provide 6600 eKW of high quality power. The prime object in the design of this complex is to provide an uninterrupted source of power. Therefore, continuity of power generation on changeover from reactor to standby power is provided for in the form of steam accumulators. The layout of the proposed power complex is shown in Dwgs. M11594-6 and M11594-7. A complete description of the complex can be found in APAE 68.[4]

The SM-2 core is made up of 45 stainless steel cald plate-type fuel elements similar to those in the SM-1 and is controlled by 7 SM-1 type Eu_2O_3 control rods in an open array. The core is reflected by laminated stainless steel which also acts as thermal shielding and is contained in a stainless steel pressure vessel. A detailed description of the core and vessel can be found in APAE 69.[5]

The major primary system components are the pressure vessel with its contained core and associated primary shielding, a vertical steam generator, a pressurizer, a single-canned-rotor circulating pump, and interconnecting piping. All primary system components that are in a vertical, cylindrical vapor container 20 ft in diameter. Adjacent to the vapor container is the spent fuel pit with a fuel transfer tube connecting it with the vapor container. The layout of the vapor container and spent fuel pit is shown in Dwg. M11594-56.

14.2.3 General Design Philosophy

The shield design of the SM-2 reflects the stringent requirements of the power complex in which it will be used. If the reactor has to be shut down to repair or adjust a component within the vapor container, enough primary shielding has to be employed so that the vapor container can be entered within a few hours after shutdown, therefore minimizing reactor down time. Sufficient secondary shielding is placed around the vapor container so that the dose rate in the remainder of the power complex is at or below working tolerance levels if an accident occurs which releases fission products or other activity within the vapor container of one of the reactors. If such an accident occurs, free access to the remainder of the complex must be maintained so that the system can produce an uninterrupted supply of power. The dose rate levels in the plant have to be sufficiently low so that a small staff can operate and maintain the system without any of the personnel receiving more than the maximum permissible dose rate of 300 mr per week. Particular attention is paid in this design to minimizing any hot spots by employing local shielding around penetrations of the primary shield and shielding any areas which may contribute to the dose rate by scattering around the shields.

14.3 SHIELD DESIGN DATA SUMMARY

In the tables below are listed the dimensions and other design parameters of the shielding required for the SM-2 plant.

14.3.1 Primary Shield

Region	Material	Outer Radius (In.)	Thickness (In.)
Core	-	11.195	
Water gap	Water	11.283	0.0880
Thermal shield	Stainless steel	12.283	1.000
Water gap	Water	12.533	0.25
Thermal shield	Stainless steel	13.658	1.125
Water gap	Water	13.908	0.25
Thermal shield	Stainless steel	16.283	2.375
Water gap	Water	16.533	0.25
Thermal shield	Stainless steel	19.721	3.1875
Water gap	Water	20.502	0.7813
Pressure vessel	Stainless steel	20.002	4.50
Insulation	Void (assumed)	28.672	3.625
P.V. support ring	Carbon steel	30.127	1.5
Water annulus	Water	53.127	23
Shield tank	Carbon steel	53.627	0.5

Region	Material	Outer Radius (In.)	Thickness (In.)
Heavy Shield Lead, ρ= 11.3 or Concrete ρ= 2.4		61.127 86.127	7.5 32.5

14.3.2 Secondary Shield

Material	Naturally occuring earth, sand or gravel

14.3.3 Spent Fuel Shipping Casks

Shielding material	Lead
Elements per cask	6
Thickness, in.	10-3/4
Weight, tons	8.6
Cooling time before shipment (days)	120
Dose rate one meter from cask - mr/hr	10
Decay heat, watts/element	280
Deposited in elements	80%
Deposited in shield	20%

14.3.4 Demineralizer Cask Shielding

Material	Lead
Thickness, in.	4.4
Dose rate one meter from surface after cooling time of 24 hr (mr/hr)	10

14.3.5 Hot Waste Tank

Active source	(See Section 1.2)
Material	Portland concrete
Sheld thickness, ft.	2.4
Dose rate on surface of shield (mr/hr)	(See Section 1.2)

14.3.6 Radiation Dose Rates

	Design Dose Rate (mr/hr)	Calculated (mr/hr)
Normal operation		
Adjacent to secondary sheidling	0.75	
Dose rate outside the primary shield		
2.5 hrs after shutdown, mr/hr	20	12-16
Dose rate during fuel transfer		
Integrated dose (mr/wk)	300	
Maximum dose rate	100	
Dose rate above spent fuel pit		
15 hr after shutdown	7.5	

14.4 PRIMARY SHIELDING

The primary shield of the SM-2 is designed to protect personnel within the vapor container during maintenance and refueling operations. Shielding of the spent fuel pit and shipping casks is considered with the primary shielding since the source is similar and the water level in the spent fuel pit is determined by the water level in the upper shield tank.

14.4.1 Primary Shield Design Criteria

Primary shielding design requirements for the SM-2 are based principally on the shutdown dose rates. At 2.5 hrs after shutdown, following any reactor operation time, the dose rate within the vapor container is restricted to less than 20 mr/hr except during fuel transfer. This dose rate is based on the number of hours generally required for maintenance purposes inside the vapor container and a permissible weekly dose of 300 mr. While the fuel is being transferred, a maximum of 100 mr/hr is allowed at the operator positions above the upper shield tank surface and above the spent fuel handling tank. The spent fuel shipping casks are shielded to meet I.C.C. shipping requirements when they contain spent fuel that has been cooled for 120 days.

Precautions are taken to assure that the shields are adequately cooled to prevent overheating from deposition of radiation energy. Also, the design has been examined for integrity to assure that no cracks or ducts are allowing localized hot spots to occur.

14.4.2 Radial Dimensions at Core Midplane

As part of the design of the SM-2 primary shield, a survey was undertaken to determine the material thickness and arrangements that would result in a dose rate of 20 mr/hr on the shield surface 2.5 hr after shutdown.[3] The shield configurations studied consisted of water annuli of varying thickness followed by a dense shield of concrete, load, or lead shot in water. Concrete provided the most economical shield while the lead shield was the thinnest. Because of space limitations, it was necessary to use the lead shield on that portion of the primary shield circumference facing the other primary system components which consist of the steam generator, primary pump, pressurizer and interconnecting piping. The remainder of the primary shield is made of concrete.

14.4.3 Radiation Distribution in Core Midplane

The SM-2 radial shielding in the core midplane consists of a 23 in. water annulus followed by either concrete or lead. The reactor vessel is located near the edge of the vapor container with concrete shielding used on the sides and between the shield tank and the vapor container. Toward the center of the vapor container, lead is utilized on a 120° sector of the shield to provide a thinner shield and hence allow more space within the vapor container.

Neutron flux distributions and operating and shutdown gamma dose rates were calculated in both the lead shielded and the concrete shielded sectors of the reactor.

14.4.4 Neutron Fluxes

Neutron fluxes were calculated utilizing the Valprod[6] IBM 650 code, a modified two-group multiregion diffusion code. Previous experience with this code[7] has indicated that the results are conservative[8]. Table 14-1 tabulates the material constants used in the calculations.

The radial neutron flux distributions shown in Figs. 14-1, 14-2, and 14-3 resulted from the code. Figure 14-1 shows the distribution from the core to the outside of the shield tank while Figs. 14-2 and 14-3 show the fluxes in the lead and concrete respectively.

These neutron fluxes were used as input for the ROC codes[7] for use in the secondary gamma production calculations. In addition, they were used to determine the neutron flux levels at the instrument well locations.

TABLE 14-1
VALPROD MATERIAL CONSTANTS

		510°F			150°F(10)			
	Core	Water	Steel	Insulation	Water	Steel	Concrete(17)	Lead
Dth	0.245104	0.297148	0.326055	9.99999	0.143400	0.303789	0.6000	1.039
D_f	1.625451	2.015833	1.314459	9.99999	1.6758612	1.227270	0.8191	3.60
Σ^a_{th}	0.328997	0.009947	0.144415	0-----0	0.112226	0.219076	0.01547	0.00478
τ	50.166205	54.253082	161.635806	9.99999	34.842833	150.3994	350.0000	104.5000
P	0.570555	0.988621	0.000004	1.00000	0.9296183	0-----0	0.9490	0.99532
K_f	1.267917	0------0	0------0	0-----0	0-------0	0-----0	0----0	0-----0
K_{th}	1.478983	0------0	0------0	0-----0	0-------0	0-----0	0----0	0-----0

14.4.5 Operating Gamma Dose Rates

Gamma dose rate distributions were calculated utilizing the ROC codes which employ 5 gamma ray energy groups:

Energy Group	Photon Energy (Mev)	Average Energy (Mev)
1	7	7
2	5-7	6
3	3-5	4
4	1-3	2
5	0-1	0.75

The ROC codes calculate gamma dose rates within a shield due to gamma rays arising from neutron interactions in the shield and due to gamma rays from the core. The ROC codes calculate gamma dose rates within a shield due to gamma rays arising from neutron interactions in the shield and due to gamma rays from the core. The ROC codes are based upon exponential attentation of the gamma rays with infinite medium Moments Method buildup factors following a treatment similar to that given by Rockwell.[9] Experience with this code[7] has indicated that it produces results which are conservative.

The results of the ROC codes for operating gammas are itemized in Tables 14-2 and 14-3 and are plotted in Figs. 14-4 and 14-5 for the lead and concrete sectors of the shield respectively. Gammas arising from neutron interations in the shield and from the core are listed separately in the tables and the total dose rates are plotted in the figures. It can be seen that the operating dose rate at the outside surface of the 7.5 in. lead shield is 7.5 r/hr, while the dose rate outside of the 32.5 in. concrete shield is 3.2×10^2 r/hr. These leakage dose rates are used in section 3.3.2 as a source for evaluation of the secondary shielding.

14.4.6 Shutdown Gamma Dose Rates

The ROC codes[7] were used again to calculate the dose rates through the lead and concrete shields 2.5 hr after shutdown. The data calculated are presented in Table 14-4 and 14-5 and in Figs. 14-6 and 14-7. It may be seen that the shutdown dose rate outside of the lead sector is 12 mr/hr and outside the concrete sector 15.3 mr/hr. The principal source for these dose rates is activation of the thermal shielding materials.

TABLE 14-2
OPERATING GAMMA DOSE RATES THROUGH LEAD SECTOR

Inches from Core Center		Gamma Dose Rate from Neutron Interactions in Shield (r/hr)	Gamma Dose Rate from Core (r/hr)	Total Gamma Dose Rate (r/hr)
30.13)	Water	6.60×10^5	3.57×10^4	6.95×10^5
38.75)	Water	2.66×10^5	1.68×10^4	2.83×10^5
47.38)	Water	1.24×10^5	7.10×10^3	1.31×10^5
53.13)	Water	7.68×10^4	4.14×10^3	3.10×10^4
53.63)	Shield Tank	5.05×10^3	3.86×10^2	5.43×10^3
55.50)	Lead	5.05×10^3	3.86×10^2	5.43×10^3
58.31)	Lead	1.73×10^2	1.75×10^1	1.90×10^2
60.19)	Lead	1.97×10^1	2.14×10^0	2.19×10^1
61.12)	Lead	6.77×10^0	7.53×10^{-1}	7.52×10^0

TABLE 14-3
OPERATING GAMMA DOSE RATES THROUGH CONCRETE SECTOR

Inches from Core Center		Gamma Dose Rate from Neutron Interactions in Shield (r/hr)	Gamma Dose Rate from Core (r/hr)	Total Gamma Dose Rate (r/hr)
30.13)	Water	6.59×10^5	3.57×10^4	6.95×10^5
41.63)	Water	2.04×10^5	1.28×10^4	2.17×10^5
53.13)	Water	7.68×10^4	4.13×10^3	8.09×10^4
53.63)	Shield Tank	5.69×10^4	3.14×10^3	6.00×10^4
71.68)	Concrete	3.17×10^3	1.32×10^2	3.30×10^3
86.13)	Concrete	3.08×10^2	1.50×10^1	3.23×10^2

TABLE 14-4
SHUTDOWN GAMMA DOSE RATES THROUGH LEAD

Inches from Core Center	Gamma Dose Rate Due to Activation of Shield Materials (r/hr)	Gamma Dose Rate Due to Activation Gammas from the Core (r/hr)	Total Gamma Dose Rate (r/hr)
30.13)	5.21×10^{3}	6.11×10^{1}	5.27×10^{3}
38.75) Water	1.62×10^{3}	2.69×10^{1}	1.64×10^{3}
47.38)	4.58×10^{2}	1.27×10^{1}	4.59×10^{2}
53.13)	1.97×10^{2}	6.18×10^{0}	2.03×10^{2}
53.63) Shield Tank	1.36×10^{2}	4.52×10^{0}	1.40×10^{2}
55.5)	1.29×10^{1}	6.35×10^{-1}	1.36×10^{1}
58.31)	3.98×10^{-1}	2.29×10^{-2}	4.20×10^{-1}
60.19) Lead	3.75×10^{-2}	2.33×10^{-3}	3.98×10^{-2}
61.12)	15×10^{-2}	6.91×10^{-4}	1.21×10^{-2}

TABLE 14-5
SHUTDOWN GAMMA DOSE RATE THROUGH CONCRETE SHIELD

Inches from Core Center	Gamma Dose Rate Due to Activation of Shield Materials (r/hr)	Gamma Dose Rate Due to Activation Gammas from the Core (r/hr)	Total Gamma Dose Rate (r/hr)
30.13)	5.21×10^{3}	6.11×10^{1}	5.27×10^{3}
41.63) Water	1.07×10^{3}	2.31×10^{1}	1.09×10^{3}
53.13)	1.97×10^{2}	6.18×10^{0}	2.03×10^{2}
53.63) Shield Tank	1.36×10^{2}	4.51×10^{0}	1.39×10^{2}
71.68) Concrete	1.21×10^{0}	-	1.21×10^{0}
86.12)	1.53×10^{-2}	-	1.53×10^{2}

14.4.7 Shield Tailoring

The shielding specified in the core midplane is more than adequate for points above the midplane. On the other hand, the shielding specified assumes no discontinuities in the shielding. A material savings will result from tailoring the lead primary shielding by making it thinner where slant penetration through the water and lead provide greater shielding than that available in the core midplane. However, patching of the shielding will be necessary in the vicinity of penetrations of the shield.

14.4.7.1 Lead Thickness as a Function of Height

As the elevation of a dose point above the core midplane increases, several effects take place. Increased water thickness due to slant penetration provides additional attenuation. The increased distance from sources in the core and thermal shield reduced the dose. The slant penetration angles through the pressure vessel increase. Greater thicknesses of steel block the radiation at the flange below the vessel head, and the slant penetration thickness of the lead primary shielding increases with the secant of the angle between the radiation path and the horizontal plane.

The effect of increased water thickness upon the dose rate as the radiation passes through higher elevations may be seen in Fig. 14-8, which shows the water attenuation of the shutdown core gammas for an integrated spectrum 2-1/2 hours after shutdown. The centerline of the fuel chute is used as a reference for water attenuation.

As a result of these plant penetration effects, the lead may be tapered. While a continuous variation in lead thickness could have been provided, a stepped shield as shown provides for greater ease of fabricating using either lead brick or lead rings.

The approach used in calculating the lead thickness required for elevations above and below the core were based upon Peebles' technique[10] for radiation obliquety incident upon slab absorbers. Peebles has developed buildup factors for various angles of oblique incidence. These buildup factors take into consideration multiple scattering. These factors require a greater shield thickness than a calculation based upon line-of-sight slant penetration thicknesses alone because scattering events may allow radiation to escape from the shield after traversing less shielding than the uncollided radiation would have traversed. This buildup factor is defined as

$$B_E = e^{\mu_o t} \frac{(\text{Transmitted})}{\text{Incident}}$$

where t is the slab thickness measured along the normal. These buildup factors are plotted in Fig. 14-9.

A concrete footing was required to support the lead. Fig. 14-10 shows total dose rate vs. concrete thickness for different times after shutdown. From these curves it was determined that 24 in. of concrete provides adequate attenuation at the base of the primary shield for both uncollided and scattered radiation.

One inch of additional lead shielding is placed around the top four ft of primary shielding to reduce the dose rate during fuel transfer operations to tolerance level. This additional tailoring of the primary shield is based upon considerations of one fuel element of source strength 3.1 x 10^{16} mev/sec at its highest position in the upper shield tank.

14.4.7.2 Streaming Through Primary Piping Penetrations

There is a radiation streaming path where the primary piping penetrates the lead primary shield and the pressure vessel wall that is potentially a severe radiation hazard. The radiation shielding by the 4-1/2 in. steel pressure vessel wall and the 7-1/2 in. of primary lead shield are absent at these penetrations. Therefore, additional shielding must be provided to attenuate the gamma streaming dose rate to biological tolerance level at shutdown.

Due to the complex geometry of the arrangement, a precise calculation of the dose rate at the penetrations cannot be performed. However, the shielding requirements can be determined by indirect methods.

At worst, the shielding required would be equivalent to that needed at the mid-plane of the reactor core. This would mean a patch that would be equivalent to the shielding provided by 7-1/2 in. of lead and 4-1/2 in. of steel. But gamma rays streaming through the annulus must either pass through the pipe at the elbow or through the pipe nozzles. The lesser of these two is the pipe elbow which would provide 1-1/2 in. of steel for gamma attenuation. The reduced density water at the elbow provides additional attenuating material which is equivalent to 1/2 in. of steel. The net loss of shielding material for this case is therefore 2-1/2 in. of steel and 7-1/2 in. of lead. Fifteen in. of lead shot with a 40 percent void factor around the elbow is therefore provided for the shield patch.

The patch is extended beyond the diameter of the duct annuli so that there is no possible direct radiation path from any region of the core without traversing the equivalent of the shielding along the midplane of the reactor core.

The steel jacket which contains the lead shot has not been considered in the calculations but will provide compensatory shielding should any bridging voids develop in the lead shot patch.

In order to assure that the above reasoning provides adequate shielding for the penetrations a calculation was performed using a geometry which is both simpler than the actual case for analysis purpose, (Dwg. C9-46-123) and more severe than the actual case from the shielding point of view. This calculation indicates, because of its severity, the amount of shielding which constitutes a maximum upper limit for the actual case.

The equation used was:

$$S = \frac{Sv}{2\pi \mu_s L^2 C} \left[\left(\cos^{-1} \frac{r_1}{R}\right) (2R^2 - r_1^2) - r_1 \sqrt{R^2 - r_1^2} \right] \frac{\sqrt{\frac{r_2}{r_o}}}{\frac{r_2}{r_o} - 1}$$

where

S_v = Volumetric source strength (mev/cm^3-sec)

μ_s = Linear absorption coefficient of source material (cm^{-1})

L = Length of annular duct (cm)

r_1 = Radius of the pipe (cm)

R = Outer radius of annulus (cm)

r_2 = Radius of source (cm)

r_o = Distance from axis of source to measuring point (cm)

C = Flux to dose converstion factor ($\frac{mev}{cm^2\text{-sec/r/hr}}$)

This equation was derived from that given by Rockwell(9) for an annular duct perpendicular to a sperical source. The equation was altered to describe a volumetric source with cylindrical geometry by the term

$$\sqrt{\frac{r_2}{r_o}} \Bigg/ \frac{r_2}{r_o} - 1$$

The input values for the calculations were:

S_v = 1.01×10^{13} mev/cm^3 - sec

μ_s = .460 cm^{-1}

L = 101.7 cm

C = 3.6×10^5 $\frac{mev}{cm^2}$ -sec/r/hr

r_1 = 17.8 cm

R = 28 cm

r_2 = 101.7 cm

The dose rate at the end of the annular duct as calculated by the above method was 4.8×10^6 r/hr. The amount of patch required to attenuate this dose to tolerance level is 16 in. of lead shot.

The gamma streaming from the annular ducts in the SM-2 will be at least a factor of 10 lower than that of the case calculated due to the SM-2 geometry, which permits line of sight streaming only for the outer edges of the core and thermal shields. All other gamma rays streaming from the ducts must pass through more shielding material (pressure vessel wall, primary lead shield, pipe walls, etc.) than that previously considered.

The difference between the shield patch provided in the SM-2, and that required by the hypothetical case examined above, is only 1 in. of lead shot. In view of the extreme differences in the geometry of the two cases, the 15 in. of patch provided in the SM-2 design is conservative.

Scattering provides additional problems. Where the radiation path through lead passes near the surface of the shielding, scattering events may bring the scattered gammas out of the shield through a shorter path length than is afforded by a line of sight path from the source.

The energy degradation in Compton scattering, which is the predominant scattering mode for the energies involved here, is given by:

$$\frac{E}{E_o} = \frac{1}{1 + \frac{E_o}{0.51}\,(1 - \cos\theta)}$$

where

E = energy of the scatter photon

E_o = energy of the incident photon

θ = scattering angle

The following tabulation shows the scattered photon energy for the range of incident photon energies typical of the shutdown core spectrum hardened by passing through a portion of the primary shield and for various scattering angles.

SCATTERED PHOTON ENERGY RATIO, E/E_0

Incident Photon Energy:	1.0 Mev.	1.5 Mev.	2.0 Mev.
Scattering Angle			
60°	0.50	0.42	0.34
70°	0.43	0.35	0.28
80°	0.38	0.30	0.23
90°	0.34	0.22	0.20
100°	0.30	0.24	0.18

The ratios of the scattering to total cross sections for lead are:

ENERGY	
1.0 Mev.	0.69
1.5 Mev.	0.75
2.0 Mev.	0.74

In those portions of the shielding around the piping where scattering was likely to be a problem, shielding was added to eliminate the problem. A typical calculation is outlined here to illustrate the method used.

The inlet pipe to the pressure vessel forms an angle of 15° with the normal to the primary shield water tank. Primary radiation from an assumed source at the central portion of the core cannot intersect the closest wall of the pipe. To escape the shield, most of the core radiation will pass essentially parallel to this wall. However, radiation from a significant portion of the core and thermal shields may pass into the pipe and be scattered by the pipe wall and the lead shielding around the pipe. The scattered radiation from the pipe wall and inner portion of the lead shielding may pass through greatly reduced material thickness where the scattering angle is from 70° to 100°. The average energy of the scattered radiation is 0.3 E_0. A reasonable conservative incident energy for this case is 1.5 Mev. The scattered ray, therefore, has an average of 0.45 Mev. The μ/ρ for lead at this energy is 0.2.

The 1.5 mev gammas, with a μ/ρ of 0.052, would require 15 in of lead shot to attenuate the radiation, but at the reduced energy level, only 3.9 in. would be required. Based upon the ratio of the cross sections $\left[\sum_s/\sum_T = 0.75\right]$, no more than 75 percent of this radiation may scatter before being absorbed. In addition, less than half the projected volume of the core can contribute to this scattered dose, since the primary radiation from most of the core and thermal shields cannot intersect the pipe wall under consideration. These two factors, $.5 \times .75 = .375$, are equal to the attenuation of an inch of lead shot, thus reducing the required thickness to absorb the scattered radiation to 3 in. in this region. The final shape has been made more regular than the minimum called for by the calculation to permit easier fabrication.

14.4.7.3 Upper Shield Tank

The upper shield tank is designed to provide shielding for personnel performing fuel element transfer operations. In addition, it is desired that the dose rate above the upper shield tank with all of the fuel in the coil be no greater than 7.5 mr/hr.

14.4.7.4 Dose Rate During Fuel Transfer

The dose rate during fuel element transfer was limited to a maximum of 100 mr/hr. The maximum dose rate will occur when the hottest fuel element is in the highest position. An exposure of 3 hr would be required at this dose rate to accumulate the allowed weekly dose of 300 mr. This would allow an average handling time for each element of about 5 min in the highest position which is considerably longer than the time which should be required.

It was assumed that the fuel transfer operation would take place at 15 hr after shutdown. It may also be assumed that the hottest element is 1.5 times as active as the average element.

The gamma rays were divided into four energy groups for these calculations:

Group I	0-0.25 mev
Group II	0.26 - 1.00 mev
Group III	1.01 - 1.70 mev
Group IV	$>$1.70 mev

A single spent fuel element can be assumed to be a line source. The appropriate line source equation is [11]

$$\phi = \frac{B\,S_L}{2\pi a\,C_E}\,F(\theta, b)$$

where:

ϕ = dose rate (mr/hr)

B = Buildup factor

S_L = Source strength line source (mev/cm^{-1} -sec^{-1})

C_E = Flux to dose conversion factor (mev/cm^2 sec to mr/hr)

$F(\theta, b)$ = F factor = $\int_0^{\theta} e^{-b \sec\theta'}\, d\theta'$

a = distance from dose point to line source (cm)

Source strengths were obtained from a plot (Fig. 14-11) of gamma activity vs. time after shutdown from 1 to 104 hours. These data were obtained from ORNL 2127[12] for an operating time of 2.9×10^7 sec, a total U^{235} inventory[13] of 9.605×10^{25} original atoms, a core vaolume of 1.384×10^5 cm^3 and a thermal neutron flux of 3×10^{13} n/cm^2-sec.

The dose rates calculated by the above equation for water thicknesses from 3 ft to 100 ft over one fuel element are plotted in Fig. 14-12. The dose rate requirements will be met if 8.7 ft of water is kept over the fuel elements at all times during the transfer operation. The curve also gives the dose rate to be expected if the fuel element is placed at some intermediate position.

14.4.7.5 Dose Rate from Shutdown Core

The water level in the upper shield tank should be high enough to reduce the dose rate from the shutdown core to less than 7.5 mr/hr at 2.5 hr after shutdown.

The reactor core was used as a volumetric source and the dose rate above the water was calculated. The appropriate equation for the dose rate at the end of a cylinder is

$$\phi = \frac{BS_V}{2\mu_s C_E}\left[E_2(b_1) - E_2(b_3) + \frac{E_2(b_3 \sec\theta)}{\sec\theta} \; \frac{E_2(b_1 \sec\theta)}{\sec\theta}\right]$$

where:

μ_s	= Macroscopic cross section of source material (cm^{-1})
S_V	= Volumetric source strength (mev/cm^3-sec)
b_1	$= \sum_1^n \mu_i t_i$
b_3	$= b_1 + \;_s h$
h	= height of cylinder (cm)
$E_2(b)$	$= b \int^{\infty} \frac{e^{-t}}{t^2}\, dt$

Fig. 14-13 is a plot of the gamma dose rate at the water surface versus water height above the core. It may be seen that 12 ft of water are needed to reduce the dose rate to 7.5 mr/hr.

14.4.7.6 Spent Fuel Handling System

At the end of useful core life, the spent fuel elements are removed from the reactor core and transferred to the spent fuel pit. In this pit, the fuel elements will be stored about 120 days and will then be loaded into shipping casks for shipment to the processing plant.

14.4.7.7 Fuel Element Handling

During the fuel element transfer operation, the water level in the spent fuel pit will be the same as that in the upper shield tank. The fuel element transfer tube is oriented horizontally and the handling operations to remove the fuel element from the transfer device should not result in a higher position than that required to load the fuel element in the transfer carrier. Therefore, the dose rate above the spent fuel handling pit will not exceed 100 mr/hr during the fuel transfer operation.

While loading the fuel elements into shipping casks, the elements will have to be lifted high enough to clear the sides of the cask. Fig. 14-12 gives the dose rates as a function of water height over one fuel element, and has been used to determine the shielding required to permit this operation.

14.4.7.8 Water Above Stored Core

While the core is stored inthe spent fuel pit, sufficient shielding in the form of water over the core is provided to reduce the dose rate at the water surface to 7.5 mr/hr. Dose rates from the stored core 15 hr after shutdown were calculated. The results, shown in Fig. 14-12, indicated that 12 ft of water provides shielding adequate to meet the dose rate restriction.

14.5 SPENT FUEL SHIPPING CASK

The SM-1 fuel elements have been designed with minimum sized end boxes, and may be accommodated by smaller shipping casks than those used for the SM-1 or PM-2A. Also, the lead shielding of these previous casks is inadequate to supply the radiation protection needed for the SM-2 fuel elements. A shipping cask must reduce the dose rate to shipping tolerances, provide adequate cooling for after-heat removal, and be designed such that no critical assembly of fuel could be shipped in it. Economic advantages may be accrued by designing a cask of such weight that it may be handled at the facilities where it is used and transported by all of the means required to take it to and from these locations.

To facilitate development of the most convenient and economical cask which will meet these requirements, a set of curves was developed. These curves give the minimum lead shielding thickness required, weight of each cask, and total shipping weight per core. Each of these parameters is plotted vs the number of fuel elements per cask. The resulting curves may be used as a guide for determining the number of fuel elements which may be most economically shipped in one cask without exceeding handling limitations.

The weight curves presented here are approximate and were used only as a guide for sizing and designing the casks. When the number of elements to be shipped in one cask is established and the designs are completed, they should be reviewed to assure that the shield is a good design without needlessly excessive shielding.

Assumptions which were necessary in order to develop the data desired were:

1. The casks will be designed for central fuel elements having 1.5 times the average activity.

2. The reactor has been operated for one year at 28 MW constant power.

3. The fuel has been allowed to cool 120 days.

4. The source is concentrated at a point and no self-attenuation is allowed.

5. The dose rate one meter from the cask surface is limited to 10 mr/hr.

6. The only source of activity is the fission products.

Gamma ray energies for the reactor history assumed (2 and 3 above) were taken from Reference 1 and grouped into two groups. Group A was used to represent those gamma rays having energies from 0 to 1.2 Mev, while group B represented those of energies greater than 1.2 Mev. Average energies of 0.75 and 1.7 Mev respectively were used since most of the activities are near these energies. The rate of gamma emission per fuel element was found to be:

Energy Group	Source Strength (Mev/sec)
A	5.47×10^{14}
B	1.03×10^{13}

Dose rate from one element, $D_{(1)}$, as a function of lead shielding thickness was then calculated from the following equation:

$$D_{(1)} = \frac{C_{(E)}\, B\,(E, \mu X)\, S e^{-\mu X}}{4 \pi r^2}$$

where

B = Buildup factor

S = Source strength (Mev/sec).

$C_{(E)}$ = Flux to dose conversion factor $\frac{R/hr}{Mev/sec}$

μ = Gamma ray linear absorption coefficient $\left(\frac{1}{cm}\right)$, and

X = Lead thickness (cm).

The curve which was plotted from these data allowed the minimum lead thickness required to reduce the dose rate from any number of fuel elements to 10 mr/hr at 1 meter[14] to be determined. This procedure resulted in Fig. 14-14, which gives the lead thickness required for the shipping casks as a function of the number of elements contained therein.

Weight calculations were based on lead casks of sufficient length to accomodate fuel elements 26-1/4 inches long and having a 3-3/4 inch square cross sectional modulus. The separation between the element and the inside of the cask to allow for cooling and structure was taken to be 0.75 in., the same as for the PM-2A[15] shipping casks. Lead thicknesses taken from Fig. 14-14 were used with simplified geometrical models of the required lead volumes to calculate the approximate cask weights. The weights obtained are presented in Fig. 14-15. The total lead shielding weight required to ship 45 fuel elements is also shown. The economic advantage of using larger casks is evidenced in this figure.

Provisions must be made in the shipping cask for the removal of after-heat. The amount of fission product power deposited in the core was taken from data presented by Clancy and Stehn[17]. This is 280 watts or 956 Btu/hr for each fuel element. The heating load as a function of the number of fuel elements is presented in Fig. 14-16.

After considering the parameters presented here and the convenience of small shipping casks, a cask for 6 fuel elements was designed (Dwg. M11594-95). This cask is similar in design to the SM-1 cask, with reduced length and increased lead thickness.

14.6 INSTRUMENT WELL SHIELDING

The nuclear instrumentation for the SM-2 must be capable of supporting a reactor startup in no more than 1 hr after a hot scram. In order for the instruments to give accurate readings at the lower limits of their ranges, the gamma dose rate must not exceed a maximum level of 10^3 R/hr for BF_3 instruments and 33 R/hr for compensated ion chambers.

From Fig. 14-6, the gamma dose rate at the instrument well position 2.5 hr after shutdown is 5.2×10^3 R/hr. This value was used with the values of the gamma dose rate as a function of time after shutdown for the SM-1 which are presented in Fig. 2.3 of APAE No. 35 [8] to find the gamma dose rate one hour after shutdown. This was 10^4 R/hr and includes both fission product gammas and activation gammas.

The range of the detectors and the maximum permissible gamma background at the lower limit of the range are listed below:

Instrument	Range (NV)	Max Gamma Background (R/hr)
BF_3	$2.5 (10^{-1}) - 2.5 (10^4)$	10^3
Compensated Ion Chamber	$2.5 (10^2) - 2.5 (10^{10})$	33

When startup is initiated after a hot scram, the thermal neutron flux will increase at a rate proportioned to the reactor period. The gamma flux will increase at a slower rate during startup, since it is primarily due to decay gamma radiation and may be assumed constant for a short time after startup is initiated. With this assumption, the gamma flux at the detector locations is 10^4 R/hr when the detectors give readings at their lower limits. The gamma flux must be attenuated by a factor of 10^{-1} for the BF_3 counter and 3.3×10^{-3} for the compensated ion chamber.

In order to determine the lead thickness required to attenuate the gamma flux by these factors, the gamma energy distribution at the instrument well position was assumed to be the same as the fission product gamma spectrum one hour after shutdown after 100 hr of constant power reactor operation.

The gamma ray spectrum from U-235 fission products for these conditions was obtained from reference (16).

The lead shielding required on the core side of each instrument are shown below where account has been taken of the fact that the lead shield replaces an existing water shield of the same thickness.

Instrument	Lead Shielding Thickness On Core Side of Instruments (inches)
BF_3	1.8
Compensated Ion Chamber	5.1

These lead thicknesses will reduce the thermal neutron flux by a factor of 2 for the BF_3 counter and by a factor of 8 for the compensated ion chamber.

The gamma dose rate, due to neutron capture in hydrogen and compton scattering in the water gap behind the instrument were considered for the compensated ion chamber because of its low gamma tolerance.

The contribution to the dose rate, due to neutron capture gammas from water was shown to be negligible by the following argument:

The least neutron flux that can be measured by the compensated Ion Chamber is $2.5\ (10^2)$ NV. This flux is assumed to be totally absorbed in the water gap. If each capture gives rise to a 2.2 mev gamma ray, the resulting gamma flux will be 4×10^{-4} R/hr. This is insignificant compared with the permissible dose rate of 33 R/hr.

The contribution to the dose rate, due to scattering in the water behind the instrument well, was determined by tracting the flux at the instrument well position as a parallel beam of 1 mev gamma rays. A point kernel for the dose rate at the detector from scattered gamma rays was defined and integrated over the half space behind the detector well.

The backscattered dose rate thus calculated was 1.4 percent of the uncollided dose rate or 140 R/hr.

The lead thickness required behind the compensated ion chamber to reduce the dose rate from 140 R/hr to 33 R/hr is 0.8 inches. No shielding is required behind the BF_3 counter.

14.7 AFTERHEAT

The energy produced by the fission products in a reactor has been calculated by Stehn and Clancy.(18) Virtually all of the beta energy and about 60 percent of the gamma energy is converted to heat before leaving the reactor core. The total fission product energy as a fraction of operating power is given in Fig. 14-17 as a function of time after shutdown after one year operation at 28 MW. The power will be distributed about 80 percent in the core and 20 percent in the shielding.

14.8 SHIELD HEATING

14.8.1 Gamma Heating Distribution During Operation

The gamma heating rate was calculated by the ROC code(7) during full power operation using the applicable property constants for each material region traversed. Conversion constants were caclulated by the formula

$$H_{Material} = \frac{1}{1.55 \times 10^{-8}\, \mu_e\, E}$$

where

1.55×10^{-8} = Conversion factor from Mev/cm^3 - sec to BTU/ft^3-hr

μ_e = Energy absorption coefficient for the material region, cm^{-1}

E = Energy, (MEV/photon)

$H_{Material}$ = Conversion factor from gamma flux to heat deposition

Table 14-6 gives the conversion constants for each material used and the energy absorption values used for gamma heating.

The gamma heating rates for lead and concrete are plotted for each material region and are shown in Fig. 14-18 and 14-19, respectively.

TABLE 14-6
ENERGY ABSORPTION COEFFICIENTS AND PROPERTY CONSTANTS FOR GAMMA HEATING

Energy (mev.)	Energy Absorption Coefficients (cm^{-1})			
	Water	Iron	Concrete	Lead
7	0.018	0.188	0.0534	0.4537
6	0.019	0.184	0.0514	0.4196
4	0.021	0.177	0.0507	0.3629
2	0.026	0.161	0.0572	0.3232
0.75	0.033	0.212	0.0661	0.6919

Energy (mev.)	Conversion Constants, Hg			
	Water	Iron	Concrete	Lead
7	5.128×10^{10}	4.90×10^{9}	1.727×10^{10}	2.031×10^{9}
6	5.649×10^{10}	5.84×10^{9}	2.091×10^{10}	2.56×10^{9}
4	7.692×10^{10}	9.112×10^{9}	3.180×10^{10}	4.445×10^{9}
2	1.240×10^{11}	2.004×10^{10}	5.640×10^{10}	9.981×10^{9}
0.75	2.60×10^{11}	4.045×10^{10}	1.301×10^{11}	1.243×10^{10}

14.8.2 Neutron Heating

In order to ascertain the importance of fast neutron heating relative to gamma heating in the lead and concrete shields,an estimation of the heat generation due to fast neutrons was obtained and compared to the gamma heating in these regions. It was assumed that the kinetic energy of all fast neutrons incident upon the surface of a region is abosrbed within that region, and that the average kinetic energy per neutron is 1 mev.

The rate of heat deposition was calculated for a position of each region corresponding to the 22.5 in. core. Above and below the core,the fast flux falls off rapidly. However, neutron energy incident on a ten inch section of each region both above and below the core was also calculated. In either case the heating was calculated by the equation.

$$H = \phi A \overline{E}$$

Where

H = neutron heating in mev/sec.
ϕ = average fast flux at inner surface of region.
A = inner surface area of region on which flux is incident.
$\overline{E}$ = average neutron energy - 1 mev.

Data were obtained from AP Notes 145[13] and 188[18] and from the Windowshade code. In calculating the heating in that part of each region corresponding to the 22.5 in. core, the following constants were used:

Region	Inner Radius (in.)	Incident Flux (n/cm^2 sec)
Lead +	136	1.76 (10^7)
Concrete	136	8.0 (10^5) .

A linear average of the flux at the top of the core and the flux 10 in. above the core was used in calculating the heating in the 10 in. section of each region above the core.

The average core fast flux is $\phi = 4.243 \times 10^{14}$.

Flux at top of core: 10^{14}
Flux 10" above top of core: $2.4\ (10^{12})$

taking a linear average $= \dfrac{10^{14} + 2.4\ (10^{12})}{2} = 5.12\ (10^{13})$

the ratio of the flux above core to the average core flux $= \dfrac{5.12\ (10^{13})}{4.243\ (10^{14})} = 0.1207$

Therefore, the average flux incident on a 10 inch section of the lead or concrete above the core is $(0.1207)\ (1.76 \times 10^7) = 2.12 \times 10^6$ n/cm^2 sec.

The fast neutron heating at a distance greater than 10 in. above or below the core is negligible compared to these values.

The rate of heat generation due to fast neutrons is less than that due to gamma rays by a factor of 10^3 in the lead and concrete.

Since 98 percent of the fast neutrons have energies less than 10 mev and neutrons reactions will deposit less than 10 mev, an upper limit to the neutron heating may be obtained by assuming that all neutrons incident upon the surface of a region each deposit 10 mev of energy within that region. Therefore, in the extreme case, the neutron heating will be no more than 1 percent of that due to gammas.

The neutron heating in the lead and concrete shields is negligible compared to the gamma heating and additional cooling provisions, over that necessitated by the gammas,are not required. A more refined calculation of neutron heating using the neutron reactions and their subsequent energy releases is therefore, not justified for this shielding geometry.

14.9 SECONDARY SHIELDING

14.9.1 General

The purpose of secondary shielding in the reactor complex is twofold. During normal operation the secondary shield maintains a low dose outside of the primary system area. In the event of fission products accidently being released from the reactor, the dose rate outside of the primary system area will be within tolerance levels.

The secondary shield, or vapor container, in satisfying the above requirements shall also prevent the dispersion of activated materials into the countryside of the event of an accident.

14.9.2 Design Criteria

The secondary shield of the SM-2 is designed to permit no more than 10 percent of laboratory tolerance, 0.75 mr/hr, exterior to the vapor container shield during normal full power operation. In addition, there is sufficient secondary shielding to limit the dose rate to 7.5 mr/hr outside a specified exclusion area should a maximum credible accident occur. This exclusion area is laid out so that there is free access to the remainder of the power complex.

14.9.3 Shielding Required for Maximum Credible Accident

The theoretically maximum credible accident for the design of the SM-2 vapor container has been defined in APAE 63.[19] Briefly, the max-crax consists of a simultaneous rupture of the primary and secondary systems. The core is then assumed to melt down and release a portion of its fission products in a cloud form.

For the purpose of a shielding study calculation, 20 percent of the total number of curies contained in the core are assumed to be released in the cloud. This is slightly more conservative than the experimental data which indicate that about 16 percent of the activity will be released. Other assumptions made for the calculations are that a 2 mev gamma per disintegration is produced, and point source geometry can be used to approximate the cloud source.

In the process of completing the initial design survey of the vapor container concept it was necessary to determine the secondary wall thickness needed to reduce the dose rate to 7.5 mr/hr on the outer surface of the vapor container under "max-crax" conditions. A calculation based on the following data was made for this purpose:

1. Number of atoms U-235 in SM-2 core per unit volumes - 6.94×10^{20} atoms/cm^3.

2. Core volume - 1.384×10^5 cm^3.

3. Core lifetime = 25.8 MWYR.

4. Total number of curies per original atom of U-235 = 1.9×10^{-18} curies/atoms U-235.

5. $\mu/\rho = 0.0459$.

6. $\rho = 2.4\ g/cm^3$ for ordinary concrete.

7. The source strength due to the activated nuclides in the cloud was calculated as

$$So = 2.69 \times 10^{18}\ MEV/sec$$

The following equations were used:

$$D = \frac{SBe^{-b}}{4\pi R2\ C_E} \qquad (1)$$

$$B = A_1 e^{-\alpha\mu_o X} + (1-A_1)e^{-\alpha\mu_o X} \qquad (2)$$

These equations are found in reference(11).

The possibility of using an exclusion area around the vapor container to reduce the amount of concrete necessary for secondary shielding was investigated. The following equation was used to determine the area size needed to make a significant change in secondary shield depth.

$$\frac{(\text{Radius of specified point})^2}{(\text{Radius of vapor container surface})^2} \times 7.5\ mr/hr = \text{Maximum Dose Rate for } 7.5\ mr/hr.$$

The results of these calculations are shown on Fig. 14-20 and 14-21. Reference to Fig. 14-20 shows that in order to attain a dose rate of 7.5 mr/hr at the outside of the vapor container, a concrete wall 79 in. (6.6 ft) is necessary. This would be impractical as well as expensive. Fig. 14-21 shows the relationship between the distance from the center of the vapor container and concrete wall thickness for an isodose of 7.5 mr/hr. It is readily seen that for 3 ft of concrete the exclusion area would have to be approximately one mile. This, too, is impractical. An alternate method of protection was devised by partially burying the vapor container in the ground and using a mound of dirt backfill for shielding. This solution permits the use of a material which is not only inexpensive but also available in unlimited amounts. The layout of a reactor with such a secondary shield arrangement is shown in APAE 63.(19)

14.9.4 Shielding Required During Normal Operation

14.9.4.1 Calculation of N^{16} Activity in Primary Water

A principal source of gamma activity in power reactors in which the primary coolant in water is due to the 0^{16} reaction in the water. This reaction, whose threshold is 10.2 mev, is shown in the formula:

$$_8O^{16} + {}_0n^1 \rightarrow {}_7N^{16} + {}_1p^1$$

where an oxygen nucleus captures a fast neutron, transforms into $_7N^{16}$ nucleus and decays by β^- emission with a characteristic half-life of 7.35 seconds. Fifty-five percent of these disintegrations are accompanied by a 6.13 mev gamma ray, 20 percent by a 7.13 mev gamma ray, and 1 percent by a gamma ray energy which may range from 8.89 mev to 1.76 mev. The remaining β^- decays go directly to the ground state. [20]

14.9.4.2 Cross Section for N^{16} Activation

The cross section for the $_8O^{16}$ activation as a function of energy has been investigated by Martin[21] and his results are reproduced in Fig. 14-22 with the cross section at 18 Mev increased to 60 mb in accordance with Martin's "note added in proof" to reference (21).

An average cross section, $\bar{\sigma}$, for neutrons above 10 mev distributed in energy according to Watt's fission spectrum was calculated as follows:

$$\frac{\int_{10\ \mathrm{Mev}}^{\infty} N(E)\ \sigma(E)\ (E)\ dE}{\int_{10\ \mathrm{Mev}}^{\infty} N(E)\ dE}$$

where

$$N(E) = 0.484 \sinh \sqrt{2E}\quad e^{-E} \text{ (see Ref. 22).}$$

The numerator was integrated numerically; the denominator was integrated analytically using error functions.

$$\bar{\sigma} = \frac{1.96 \times 10^{-2}}{1.205 \times 10^{-3}}$$

(Ref. 22 reports the integral of the denominator as 1.2×10^{-3})

$$= 16.3 \text{ mb}$$

14.10 ACTIVATION REGIONS

In the SM-2 reactor, the primary water will be activated in all regions within the pressure vessel except the stagnation volume in the lower end. The stagnation volume is that water which resides statically in the pressure vessel. The activating regions within the vessel are indicated by Roman numerals in the schematic of Fig. 14-23.

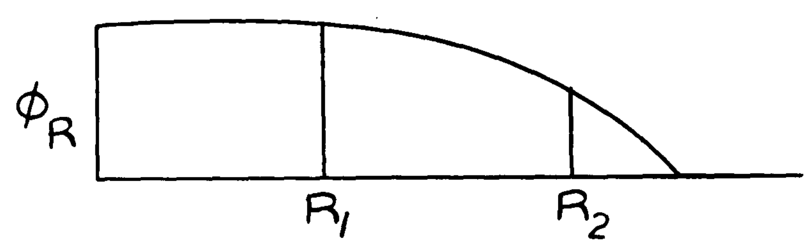
The region designated by IX is the last separate activating region according to the geometry of the system. However, as the primary coolant leaves region VIII it rises through region XX to the top of the vessel chamber and then must flow downward to pass through the exit nozzle. Thus, about one-third of it is again exposed to the activating flux in this region near the exit nozzle. For computational purposes this was designated as region X.

14.10.1 Flux Distribution in Activating Regions

To compute the average flux above 10 mev, the assumption was made that the neutron flux above 10 mev has the same radial distribution as the fast group flux. Therefore, the initial radial fast flux distribution as calculated by Valprod, an IBM650 two-group diffusion code, and shown in Fig. 14-24 will be used. The flux distribution of Fig. 14-24 has been normalized to yield a flux of 1.0 when averaged over the core. The following values for normalized average fast fluxes and peak to average ratios apply in the core region.(23) These values were calculated for conditions of equilibrium xenon zero MW years operation, and rods in.

Average fast flux in core	=	1
Average fast flux in Region VIII	=	1.278
Average fast flux in Region II	=	.763
Radial peak to average fast flux $= \left[\frac{\text{peak}}{\text{avg.}}\right]$ Radial	=	$P_R = 1.6518$
Axial peak to average fast flux $= \left[\frac{\text{peak}}{\text{avg.}}\right]$ Axial	=	$P_A = 1.452$
Overall peak to average $= P = P_A P_R$	=	2.402

The average radial fast flux in the reflector, region V, was calculated by the method explained in reference (24). The equation used in this method is shown below.



$$\phi_{avg} = \frac{\int_{R_1}^{R_2} \phi(R)\, RdR}{\int_{R_1}^{R_2} RdR} = \frac{2\pi \int_{R_1}^{R_2} \phi(R)\, RdR}{\pi\left[R_2^2 - R_1^2\right]} \qquad (1)$$

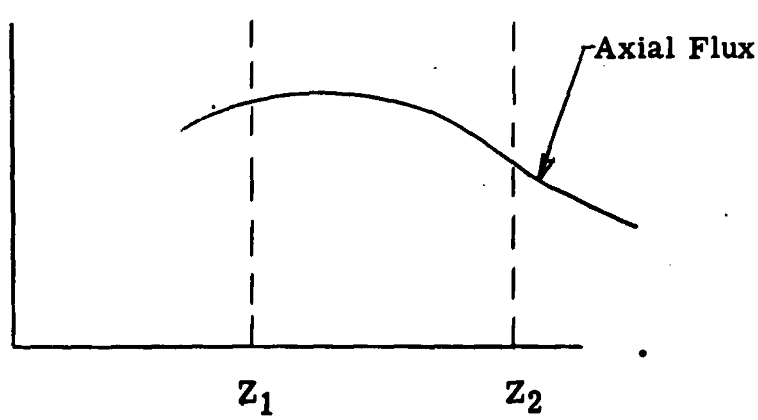
$$2\pi \int_{R_1}^{R_2} \phi(R)\, RdR = \pi(\Delta R)^2 \sum_{n=1}^{N} \phi_n \phi_{n-1} \frac{R_1}{\Delta R} + n-2 \qquad (2)$$

where

avg = average radial fast flux in the reflector region - n/cm^2-sec
R = radial distance out from centerline of core - inches.
ΔR = radial increment between flux points
$\phi(R)$ = flux at R dist. from centerline - n/cm^2-sec
ϕ_n = flux at designated points
n = index of flux points

Values for the fast flux in the reflector region were taken from Fig. 14-1. This method yeilded a peak to average flux ratio of 13. 53:1. This ratio was also used for regions IV and VI of Fig. 14-23. A radial peak to average ratio of 0. 763: 1. 6518 is used for regions I, II, III, and X and a ratio of 12. 78: 1. 6518 is used in regions VII, VIII, and IX.

The axial fast flux distribution for the SM-2 core has been calculated with Windowshade[25] (an IBM 650 code for modified two group solution of the diffusion equation in slab geometry). The axial fast flux distribution is shown in Fig. 14-25. It was necessary to calculate the axial average and peak fast fluxes in the region from the upper edge of the active core to the top inner surface of the pressure vessel. These flux distributions were established by using the method described in reference (24); the equation used in this method is given below.



$$\phi_Z = \int_{Z_1}^{Z_2} \frac{\phi(Z)\,dZ}{(Z_2 - Z_1)} \tag{3}$$

$$\int_{Z_1}^{Z_2} \phi(Z)\,dZ = \frac{\Delta t}{2} \sum_{Z_1}^{Z_2} (\phi_Z + \phi_{Z-1}) \tag{4}$$

where

ϕ_Z = relative axial flux as a function of axial mean point

Δ_Z = axial mesh spacing

ϕ_E = relative axial flux at axial mesh point

Axial flux data were taken from Windowshade and are shown in Figs. 14-26 and 14-27. An axial peak to average fast flux of 27.9:1 for the region between the lower edge of the active core and the stagnation level was determined. This axial peak to average ratio is applied to regions I, VI, and VII as described in Fig. 14-23. Similarly, the axial peak to average ratios in the region between the upper edge of the core and the top inner surface of the pressure vessel was computed to be 146:1. This axial peak to average ratio is applied to regions III, IV, IX, and X. The axial fast flux distribution in the reflector was considered to be the same as in the core. Thus from previously indicated values the axial peak to average ratios in regions II, V and VIII is 1.452:1. The following tabulation shows the axial and radial average to peak fast flux ratios.

Region	Radial (ϕ_{avg}/ϕ_{peak})	Axial (ϕ_{avg}/ϕ_{peak})
I	.763/1.6518	1/27.9
II	.763/1.6518	1/1.452
III	.763/1.6518	1/146
IV	1/13.53	1/146
V	1/13.53	1/1.4542
VI	1/13.53	1/27.9
VII	1.278/1.6518	1/27.9
VIII	1.278/1.6518	1/1.4542
IX	1.278/1.6518	1/146
X	.763/1.6518	1/146

14.10.2 Calculation of Flux Above 10 Mev

Since the variation of the flux both radially and axially has now been obtained, it is necessary to calculate the flux above 10 mev at only one point in the core to have a complete description of the flux. Therefore, the flux was calculated at the vertical centerline of the reactor core by methods outlined in references (11) and (26). (5) (33)

The calculation of the flux above 10 mev is divided into two parts: a calculation of the uncollided flux above 10 mev (ϕunc) and a calculation of the collided flux (ϕ c) above 10 mev.

14.10.2.1 Calculation of Uncollided Flux

The uncollided flux may be obtained by equating the rate of production of uncollided neutrons, S(E), to their rate of disappearance by collision, $\phi_{unc}(E)$ $\Sigma_T(e)$.

$$\phi_{unc}(E) \quad \Sigma_T(E)dE = S(E)dE \tag{5}$$

$$\phi_{unc}(E)\ dE = \frac{S(E)\ dE}{\Sigma_T(E)}$$

Where

$\Sigma_T(E)$	= total macroscopic cross section at E (cm^{-1})
E	= neutron energy (Mev)
S(E)dE	= V R_F N (E) dE (neutrons/cm^3-sec)
N(E)	= Watt's fission spectrum (neutrons/fission neutron)
N(E)	= $0.484\ e^{-E} \sinh\sqrt{2E}$
R_F	= fission rate (fission/cm^3 sec)
R_F	= $(\Sigma^f \phi)_{fast} + (\Sigma^f \phi)_{thermal}$
ν	= 2.46 for thermal neutron reactions 2.48 for fast neutron reactions (neutrons/fission)
ϕ_{fast}	= average fast neutron flux (neutrons/cm^2-sec)
$\phi_{thermal}$	= average thermal neutron flux (neutrons/cm^2-sec)
$\Sigma^f_{thermal}$	= total macroscopic fission cross section for fast neutron (cm^{-1})

νR_F $= 2.48\ (R_F)$ fast $+ 2.46\ (R_F)$ thermal

$$\text{new } \phi_{fast} = \frac{(1\text{-B})\ \delta \text{PMW}}{\Sigma^{f}_{fast}\ \ \nu_{core}} \tag{6}$$

$$\text{and } \phi_{thermal} = \frac{\text{PMW}}{\text{thermal}\ \nu_{core}} \tag{7}$$

where

B ≈ 0.00673

δ $= 3.2175 \times 10^{10}$ - fission/watt - sec

PMW $=$ reactor power $= 2.8 \times 10^{7}$ - watts

ν_{core} $=$ core volume $= 1.4 \times 10^{5}\ cm^{3}$

therefore:

$$R_F = \frac{2.48\ (1\text{-B})\ \delta\ \text{PMW}}{\nu_{core}} + \frac{2.46\ \text{B PMW}}{\nu_{core}}$$

Insertion of numerical values yields

$= 1.58795 \times 10^{13}$ neutron/cm^3-sec

$$\text{then } \phi_{unc}\ (E) = \frac{1.58795 \times 10^{13}\ (N(E))}{(E)} \tag{8}$$

Table 14-7 illustrated the uncollided flux calculation. The flux calculations were made for neutrons having energies between 10 and 17 mev. The neutron flux greater than 17 mev is negligible for the purpose of these calculations.

TABLE 14-7
UNCOLLIDED FLUX CALCULATION

E(MEV)	$\sigma_H(E)$	σ_o	σ_{FE}	$\Sigma_N(cm^{-1})$	Σ_{FE}	Σ_o	Σ_T	N(E) $x10^4$	$\frac{N(E)x10^4}{\Sigma_T}$	$\phi_{unc}(E)x10^{-9}$
10	0.94	1.25	2.95	0.03675	0.05339	0.02625	0.11639	9.63	82.739	131.385
11	0.87	1.33	2.82	0.03402	0.05104	0.02793	0.11299	4.40	38.941	61.836
12	0.79	1.41	2.68	0.03089	0.0485	0.02961	0.10900	2.00	18.348	29.135
13	0.74	1.50	2.58	0.02893	0.04669	0.0315	0.10712	0.90	8.402	13.341
14	0.69	1.54	2.52	0.02698	0.04561	0.03234	0.10493	0.40	3.812	6.053
15	0.65	1.57	2.45	0.02542	0.04434	0.03297	0.10273	0.176	7.713	2.720
16	0.61	1.59	2.39	0.02385	0.04325	0.03339	0.10049	0.078	0.776	1.232
17	0.57	1.61	2.34	0.02228	0.04325	0.03381	0.09934	0.034	0.342	0.543

1. Cross section taken from reference (27).
2. Nuclear densities taken from reference (23).
3. N(E) taken from reference (8).

14.10.2.2 Calculation of Collided Flux Above 10 Mev

The equation used to calculate the collided flux is as follows:[8]

$$\phi_c(E) = \frac{\nu R_F}{E \Sigma_H (E)} (1-P) \int_E^{\infty} N(E')\, dE' \qquad (9)$$

where

p = probability that a neutron makes its first collision inelastically with stainless steel rather than elastically with hydrogen.

$$p(E) = \frac{\displaystyle\int_E^{\infty} \frac{\Sigma_{ss}^i}{\Sigma_H + \Sigma_{ss}^i} N(E')\, dE'}{\displaystyle\int_E^{\infty} N(E')\, dE'} \qquad (10)$$

Σ_{ss}^i = stainless steel microscopic inelastic cross section (cm^{-1})

Σ_H = macroscopic elastic cross section for hydrogen (cm^{-1})

σ_{ss}^i $\approx$ 1.56 = σ_{Fe}^i

N_{ss} = number density of stainless steel = 1.81×10^{22} (atom/cm^3)

Σ_{ss}^i = .02715 cm^{-1}

Table 14-8 illustrated the calculation of the collided flux.

To calculate the total flux above 10 mev, the uncollided fluxes of Table 14-7 are added to the collided fluxes of Table 14-8 and the results integrated numerically between 10 and 17 mev. The flux at the centerline of the reactor calculated in this manner is:

$$\phi_{>10\ mev} = 2.149 \times 10^{11} \text{ neutron/cm}^2\text{-sec}$$

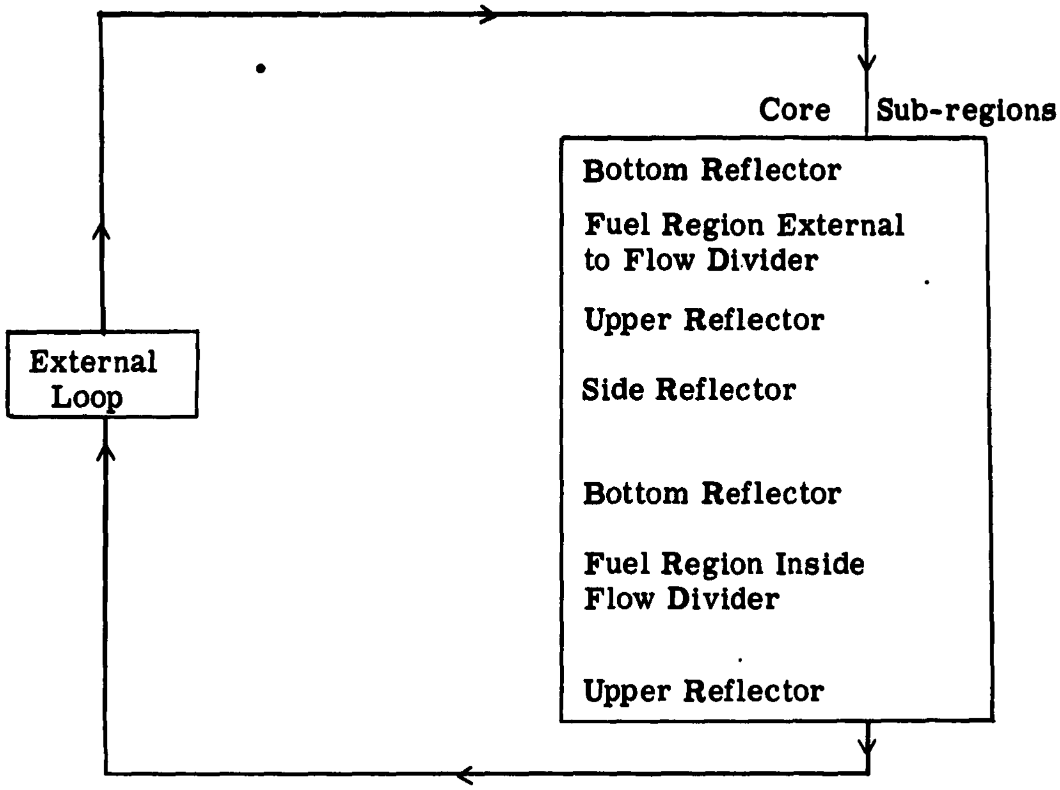
TABLE 14-8
COLLIDED FLUX CALCULATION

E (Mev)	Σ_H (cm^{-1})	$\frac{\nu R_s}{E \Sigma_H}$	$\frac{\Sigma^i_{ss}}{\Sigma_H + \Sigma_{ss}}$	N(E) x 10^{-4}	$\int_E^\infty N(E')dE'$	$N(E)\frac{\Sigma^i_{ss}}{\Sigma_H + \Sigma_{ss}} x10^4$	$\int_E^\infty N(E')\frac{\Sigma^i}{\Sigma_H + \Sigma_{ss}} dE'$	p(E)	$\frac{(1-p)\nu R_c}{E\Sigma_H} x10^{-2}$	$\phi_c(E) x10^{-3}$
10	0.03675	43.21	0.424	9.63	1.205 x 10^{-3}	4.083	5.69739 x 10^{-4}	0.4728	22.78	27.44
11	0.03402	42.23	0.444	4.40	5.45 x 10^{-4}	1.9536	2.687 x 10^{-4}	0.493	21.51	11.72
12	0.03089	42.85	0.467	2.00	2.44 x 10^{-4}	0.934	1.2446 x 10^{-4}	0.51	20.99	5.12
13	0.02893	42.22	0.484	0.90	1.09 x 10^{-4}	0.4356	0.55979 x 10^{-4}	0.513	20.56	2.24
14	0.02698	42.04	0.501	0.40	5.07 x 10^{-5}	0.2004	0.24179 x 10^{-4}	0.476	22.02	1.116
15	0.02542	41.65	0.516	0.176	2.14 x 10^{-5}	0.0908	0.09619 x 10^{-4}	0.449	22.95	0.491
16	0.02385	41.61	0.532	0.078	9.4 x 10^{-6}	0.04149	0.05079 x 10^{-4}	0.54	19.14	0.179
17	0.02228	41.93	0.549	0.034	4.08 x 10^{-6}	0.0186	0.00755 x 10^{-4}	0.185	34.17	0.139

(Microscopic inelastic cross sections taken from BNL-325 [30] and are considered constant above 10 mev.)

14.10.3 Calculation of Specific Activity of Primary Water

A schematic drawing which illustrates the SM-2 primary coolant flow scheme is shown below. This type of flow scheme is known as series flow.



The equation for the specific activity of the primary coolant as it enters the external loop is:

$$S = N(\overline{\sigma \phi})_c \, F(\lambda t_i) \left[f_c G_c \left[\lambda t_i, \frac{\lambda t_i}{\zeta}, \frac{[\sigma\phi]_i}{[\sigma\phi]_i} \right] \right] \qquad (11)$$

where

S = Equilibrium specific activity in water entering external loop (photons/cm^3-sec)

N = Then number density of target atoms in the water = IA/M

ρ = Density of water (gm/cm^3)

I = Isotopic fraction of target atoms per molecule = 0.0076 for O^{16} and 3.7×10^{-4} for O^{17}.

A = Avogadro's number = 6.023×10^{23} atoms/mole

M = Molecular weight of water = 18.02 g/mole

r,c,A = Reflector and core respectively.

The average activation probability over the range of the available spectrum above threshold in events per atom per second (where i - c or r and j = r).

$$F(\lambda t) = 1/1-e^{-\lambda t}$$

$$G_c = (1-e^{-\lambda t}) + \frac{(\sigma\phi)_r}{(\sigma\phi)_c}(1-e^{-\lambda t_r})\, e^{-\lambda t_c}$$

where

λ = Decay constant of nuclide under consideration (sec^{-1}).

T = Total cycle time (sec)

t_c = Average time a unit volume spends in the core per cycle (sec)

t_r = Average time a unit volume spends in reflector per cycle (sec)

The solution of equation (11) yields the N^{16} activity if the primary coolant as it enters the external loop as:

$$S = 1.29 \times 10^6 \text{ photons/cm}^3\text{-sec.}$$

14.10.4 Calculation of Gamma Dose Rate from Primary Coolant

The method used to calculate the gamma dose rate from the primary coolant is described fully.[28] RAS-I, an IBM 650 code developed by the Electric Boat Division of General Dynamics Corp. In this method, the primary coolant piping is divided into cylindrical sections about a foot in length and each foot-long source is approximated by a point source of strength Sv placed at the center of the section. The steam generator is divided in a similar manner. A coordinate system is introduced and the sources are located in three dimensions. The components in the vapor container (including the piping and steam generator) which afford some shadow shielding are also located in the coordinate system. The input to the computer includes data describing the piping (wall thickness, diameter) which the machine uses to approximate the self-absorption in the source. The input also includes data describing the other components (diameter, wall thickness,etc). The machine takes each point source, computes self-absorption in the pipe or steam generator, computes inverse square attenuation, checks to see if a ray from the source to the dose point passes through a component and, if so, attenuates the source accordingly, interposes four shields of specified thicknesses and uses Peebles[10] data to compute slant attenuation and buildup in the shields. The machine also sums up the dose rate at the dose point from all the sources and prints out the summation. The four dose rates at each dose point may be plotted vs. shield thickness on semi-log paper to find the shield thickness which will give a permissible dose rate on the outside of the shield.

A limitation of this calculation is that the shielding interposed between the source and the dose points must be either vertical or horizontal.

Dose rates were calculated at four locations in the vapor container. Two dose points were located at the entrances where operators and maintenance personnel will require access during power operation. These points, 3 and 4, are located at the side entrance and top hatch, respectively. Points 1 and 2 are positioned in such a way that they will be in the location of highest radiation levels. Point 1 is on the vapor container wall near the steam generator and Point 2 is on the vapor container wall near the pump. The steam generator and pump contain large volumes of coolant and are therefore strong sources of N^{16} gammas. See Figs. 14-28 and 14-29.

Table 14-9 contains a listing of the input to the RAS-I calculation. All distances used in the RAS-1 calculation must be positive; therefore, the origin of the coordinate system is outside the vapor container.

14.10.5 Presentation of Results

The results of the RAS-I Program are presented in Table 14-10 and Fig. 14-30. It is readily seen that the dose rates are a maximum at Points 1 and 2 as had been predicted in the preceding discussion. The values at the locations where personnel will require access during operation are signifcantly less than those at the maximum points for the various relaxation lengths.

14.10.6 Conclusion

It is found that 3.0 ft of concrete shielding will be required at Dose Point 3 and 2.2 ft at Dose Point 4 in order to reduce the dose due to N^{16} activity to 0.75 mr/hr at the top hatch and side entrance. Since earth fill is planned for the secondary shielding, Dose Points 1 and 2 should have 9.41 and 8.64 relaxation lengths of earth fill to provide shielding equivalent to the 4.9 and 4.5 ft of concrete indicated in Fig. 14-30.

TABLE 14-9
INPUT DATA FOR RAS-I CALCULATION, SM-2

Source Point Description

Type	ID_A*	ID_B*	X (ft)	Y (ft)	Z (ft)	C_s*
Boiler End	25	001	08.1	19.9	07.00	003.353
Boiler Tube Section	26	001	08.1	19.9	09.9	536.500
		002	08.1	19.9	10.9	039.590
		003	08.1	19.9	11.4	006.500
		004	08.1	19.9	12.4	041.970
		005	08.1	19.9	13.4	041.970
		006	08.1	19.9	14.4	041.970
		007	08.1	19.9	15.4	041.970
		008	08.1	19.9	16.4	041.970
		009	08.1	19.9	17.4	041.970
		010	08.1	19.9	18.4	041.970
		011	08.1	19.9	19.4	041.970
		012	08.1	19.9	20.4	041.970
		013	08.1	19.9	21.0	057.700
		014	08.1	19.9	22.0	057.700
Schedule 1 Pipe	33	001	06.4	14.2	11.0	008.640
	34	001	06.0	15.6	10.6	020.770
	35	001	05.9	15.6	09.7	017.100
		002	05.9	15.6	08.8	015.640
		003	05.9	15.6	07.9	015.200
	36	001	06.1	16.0	06.7	030.100
	37	001	06.5	17.0	06.2	031.910
	38	001	07.1	18.0	06.5	019.320
	28	001	09.9	19.3	06.6	614.890
	29	001	10.7	19.1	06.1	014.820
	30	001	12.0	18.6	06.1	014.750
	31	001	12.9	18.3	06.8	014.650

* ID_A and ID_B are machine identification numbers for the shadow shields and source points, respectively.

\+ C_s is representative of the strength of the source points.

TABLE 14-9 (CONT'D)

Type			ID^*_A	ID^*_B	X (ft)	Y (ft)	Z (ft)	C_S^*
			32	001	13.1	18.2	08.1	010.440
				002	13.1	18.2	09.0	013.190
				003	13.1	18.2	09.9	203.730
			39	001	11.7	16.6	11.0	009.009
			40	001	10.4	16.4	10.9	008.850
			41	001	10.2	16.4	09.9	006.610
				002	10.2	16.4	09.2	007.340
				003	10.2	16.4	08.5	007.340
			42	001	10.2	16.1	07.6	007.840
			43	001	10.1	15.2	06.9	0004.188

Plane	Dose Point Identification	Relaxation Lengths in Concrete	X (ft)	Y (ft)	Z (ft)	C_{b1}^*	C_{b2}^*
3	001	0,2	10.0	25.2	11.0	10.000	02.314
	002	5,8	10.0	25.2	11.0	00.218	00.016
2	003	0,2	17.5	22.0	11.0	10.000	02.314
	004	5,8	17.5	22.0	11.0	00.218	00.016
1	005	0,2	08.1	19.9	45.2	10.000	02.314
	006	5,8	08.1	19.9	45.2	00.218	00.016
2	007	0,2	20.0	15.3	26.7	10.000	02.314
	008	5,8	20.0	15.3	26.7	00.218	00.016

* C_{b1} and C_{b2} have been multiplied by 10^{-2} to fit input.

Therefore, dose rates calculated must be multiplied by 10^2.

TABLE 14-9 (CONT'D)

Shadow Shield Description

Type	ID_A	Radius (ft)	X_1 (ft)	Y_1 (ft)	Z_1 (ft)	OA*	X_2 (ft)	Y_2 (ft)	Z_2 (ft)
Primary Lead Shield	01	5.15	10.0	10.0	03.0	.00	10.0	10.0	11.2
	02	5.10	10.0	10.0	11.2		10.0	10.0	13.2
	03	4.94	10.0	10.0	13.2		10.0	10.0	15.9
	04	4.77	10.0	10.0	15.9		10.0	10.0	17.0
	05	4.60	10.0	10.0	17.0		10.0	10.0	18.5
Primary Concrete Shield	06	1.25	16.0	12.4	01.9		16.0	12.4	18.5
	07	1.25	04.0	12.4	01.9		04.0	12.4	18.5
	08	1.25	16.2	10.0	01.9		16.2	10.0	18.5
	09	1.25	03.7	10.0	01.9		03.7	10.0	18.5
	10	1.00	15.6	07.9	15.6		07.9	18.5	
	11	1.00	14.3	07.9	01.9		04.4	07.9	18.5
	12	0.50	14.6	06.7	01.9		14.6	06.7	18.5
	13	0.50	05.4	06.7	01.9		05.4	06.7	18.5
	14	0.25	14.0	16.2	01.9		14.0	16.2	18.5
	15	0.25	06.0	06.2	01.9		06.0	06.2	18.5
Pump	16	1.75	13.6	18.2	10.4		13.6	18.2	12.0
	17	1.75	13.6	18.2	12.0		13.6	18.2	12.7
	18	1.50	13.6	18.2	12.7		13.6	18.2	16.2
Generator Flange	19	2.96	08.1	19.9	08.5	.00	08.1	19.9	11.6
Heater Boxes On Pressurizer	20	0.21	02.2	20.2	13.0	.01	04.3	19.6	13.0
	21	0.21	02.2	19.8	13.0	.01	04.1	19.2	13.0
	22	0.21	00.9	15.7	13.0	.01	02.9	15.0	13.0
	23	0.21	00.7	15.4	13.0	.01	02.7	14.6	13.0

TABLE 14-9 (CONT'D)

Type	ID_A	Radius (ft)	X_1 (ft)	Y_1 (ft)	Z_1 (ft)	OA*	X_2 (ft)	Y_2 (ft)	Z_2 (ft)
Lower Pressurizer Boiler End	24	2.00	02.5	17.4	10.4	.04	02.5	17.1	13.5
	25	1.79	08.1	19.9	07.0	.05	08.1	19.9	10.3
Boiler Tube Section	26	1.79	08.1	19.9	10.8	.06	08.1	19.9	28.1
Upper Pressurizer	27	2.00	02.5	17.4	13.5	.18	02.5	17.4	17.5
Schedule 1 Pipe	28	0.66	09.5	19.5	07.0	.29	09.6	19.2	06.2
	29	0.66	10.3	19.2	06.2	.29	11.1	18.9	05.8
	30	0.66	11.4	18.8	05.8	.29	12.4	18.5	06.2
	31	0.66	12.6	18.4	06.3	.29	13.1	18.2	17.2
	32	0.66	13.1	18.2	07.6	.29	06.6	13.9	11.0
Schedule 0 Pipe	33	0.58	06.1	14.9	11.0	.32	06.6	13.9	11.0
	34	0.58	06.2	15.0	11.0	.32	05.7	15.7	10.2
	35	0.58	05.9	15.6	10.2	.32	05.9	15.6	07.4
	36	0.58	05.9	15.7	07.0	.32	06.2	16.3	06.4
	37	0.58	06.4	16.5	06.2	.32	06.7	17.5	06.2
	38	0.58	06.9	17.6	06.2	.32	07.3	18.4	06.8
	39	0.58	12.4	16.7	11.0	.32	11.1	16.5	11.0
	40	0.58	10.7	16.5	11.3	.32	10.0	16.2	10.6
	41	0.58	10.2	16.4	10.4	.32	10.2	16.4	08.0
	42	0.58	10.3	16.6	07.9	.32	10.1	15.7	17.2
	43	0.58	10.1	15.5	06.9	.32	10.0	15.1	06.9
	44	0.14	02.5	17.4	10.4	.76	02.5	17.4	09.2
	45	0.14	02.5	17.4	09.2	.76	02.5	15.6	09.2
	46	0.14	02.5	15.6	09.2	.76	05.2	15.6	08.6

* OA is a gamma absorption term.

TABLE 14-10
DOSE FROM N^{16} ACTIVITY CALCULATED USING IBM 650 RAS-I PROGRAM

Dose Point	Relaxation Lengths of Concrete	Calculated Full Power Dose (mr/hr)
1	0	4760.64
	2	944.05
	5	71.42
	8	4.17
2	0	3834.53
	2	657.82
	5	42.38
	8	2.26
3	0	229.08
	2	36.40
	5	2.12
	8	0.10
4	0	22.51
	2	5.22
	5	0.49
	8	0.03

14.11 DEMINERALIZER CASK

Activated nuclides, both soluble and insoluble, appear in the primary coolant due to the corrosion and activation of structural material.

In order to maintain a low concentration of these activated nuclides in the SM-2 primary coolant system, a resin filled demineralizer is placed in the primary coolant blowdown line. The demineralizer serves the dual purpose of acting as a filter for insoluble nuclides and as an ion exchange column for soluble nuclides. Because of this accumulation of activated nuclides. Because of this accumulation of activated nuclides in the demineralizer, the demineralizer itself becomes a radiation source.

Shielding must therefore be provided for the demineralizer during operation and for shipment of the demineralizer after usage. The I.C.C. regulation governing a spent demineralizer require that the dose rate at one meter from the demineralizer surface does not exceed 10 mr/hr. The shielding required to meet these specifications for an SM-2 demineralizer was determined in the following manner:

The accumulation of active nuclides in the demineralizer was examined for several conditions of blowdown rate, time on the line, and crud buildup in the primary coolant using the assumptions and procedure outlined in reference (11). Table 14-11 gives the accumulated activity in the demineralizer for the three cases examined.

TABLE 14-11
ACCUMULATED ACTIVITY, DEMINERALIZER

Case 1 — Normal blowdown, 1/2 year, 1 year crud

Nuclide	Activity
Fe^{59}	1398
Co^{58}	2670
Mn^{54}	2010
Ta^{182}	826
Co^{60}	994

Case II — Maximum blowdown, 1/2 year, 1 year crud

Nuclide	Activity
Fe^{59}	2796
Co^{58}	5350
Mn^{54}	4020
Ta^{182}	1652
Co^{60}	1988

TABLE 14-11 (CONT'D)

Case III

Normal blowdown, 1 year, 20 year crud

Nuclide	Activity
Fe^{59}	1670
Co^{58}	3750
Mn^{54}	6710
$T2^{182}$	2060
Co^{60}	19480

The gamma energy and yield for each emitter considered is shown below:

Nuclide	% Yield	Energy (mev)
Fe^{59}	2.8	0.191
	43.0	1.289
	57.0	1.098
Co^{58}	0.5	1.664
	1.6	0.865
	99.0	0.799
Mn^{54}	100	0.842
Ta^{182}	113*	1.12
Co^{60}	99	1.333
	99	1.172

* 1.12 mev is the predominant decay gamma energy and decay scheme is not well established, therefore, if we assumed that the yield was equal to the total decay gamma energy divided by 1.12.

The volumetric source strengths for each gamma energy were developed using the relationship

$$\text{Intensity} = S \times 3.7\,(10^7) \times N \times E \times \frac{1}{V}$$

where:

S = Source strength in millicuries
N = Yield fraction
E = Gamma energy
V = Volume = 5.66×10^4 cm^3

The different gammas emitted from the active nuclide were consolidated into three groups to facilitate the calculation of the dose rate. The energy groups that were used and the volumetric source strength for each group are tabulated below.

		Case I	Case II	Case III
Group 1 0.82 mev	E(mev) (cm^3 sec)	2.52 (10^6)	5.04 (10^6)	5.68 (10^6)
Group 2 1.15 mev	E "	2.01 (10^6)	4.02 (10^6)	1.72 (10^7)
Group 3	E	1.38 (10^6)	2.76 (10^6)	1.74 (10^7)

The IBM 650 shielding program(29)(30) was used to calculate the dose at one meter from the demineralizer surface. Lead thicknesses of 3, 4 and 5 in. were examined. The results are plotted in Fig. 14-31. These curves indicate that for the worst case (normal blowdown, 1 yr on line, 20 yr crud accumulation) a minimum of 4.4 in. Pb is required.

14.12 HOT WASTE TANK

Each of the reactors in the SM-2 complex will be equipped with one - 2500 gal holding tank and two-5000 gal storage tanks for radioactive liquid wastes. It was assumed that the largest source which might be stored in one of these tanks would be no more than the largest activity occurring in one of the Eu_2O_3 absorber plates. It was further assumed that sufficient shielding should be provided to reduce the dose rate from this source to 7.5 mr/hr. The shield developed under these assumptions would then be sufficient to reduce to 7.5 mr/hr the dose rate from a source equivalent to half of the fission products contained in one fuel plate at the end of life, and reduce the dose rate to about 10 Rem/hr from a source composed of all of the radioactivity present in the fission products and Eu_2O_3 in one of the reactor systems.

The waste tanks thus shielded would not limit operations at the complex due to radioactive nuclides stored after any minor accident and would not seriously restrict operation of one reactor after storage of all waste from any possible catastrophe with the other reactor.

The shielding requirements of the waste tanks were calculated for two source. The first source considered was equivalent to the maximum activity in one absorber plate. The second source was taken as 20 percent of the amount of fission products present in one fuel plate at the end of life. The activated Eu source was used as the basis for the shielding requirements for the waste tank because (a) the absorber provides a more active source, and (b) the absorber is a more creditable case because of the high solubility of Eu_2O_3 in water.

The equation used for the calculation was

$$D = \frac{SBR^2}{2(a+Z)\ VC}\ F\left[\theta\,(\mu_c t_c + \mu_s Z)\right]$$

where

D = Dose rate at surface of shielding (r/hr)

So = Source activity due to active fuel plate material (mev/sec)

B = Buildup factor.

Ro = Radius of cylindrical source volume (ft)

a = Distance from surface of tank (ft)

Z = Self-attenuation distance of source volume (ft)

V = Source volume (ft^3)

C = Flux to dose conversion factor $\frac{(mev/ft^2\text{-}sec)}{r/hr}$.

θ = Angle formed by cylinder ends and dose point

μ_c = Linear absorption coefficient of concrete (ft^{-1})

tc = Thickness of concrete (ft)

μ_s = Linear absorption coefficient of water (ft^{-1})

The assumptions used for this calculation were:

1. The waste tank is filled with water.
2. The active material is distributed uniformly throughout the waste tank volume.
3. The reactor was in operation for 20 mwyr.[25]
4. Portland concrete of density - 2.4 gm/cc was the shielding material.
5. The source gammas were 1.1 mev gammas.

The source strength (So) due to the Eu_2O_3 was determined by choosing an effective gamma energy for each Eu isotope (Eu^{152}, Eu^{154}, Eu^{155}) in the absorber at the end of life.[20] The activity of each isotope[31] was then multiplied by the effective gamma to give the source provided by each isotope. The effects of Eu^{155} are masked out by those of Eu^{152} and Eu^{154} so that it was then neglected. The source strength (So) was then taken as the sum of the individual source strengths due to Eu^{152} and Eu^{154}.

The constants for the 5000 gal waste tank in (1) had the following values:

S_o = 7.2×10^{14} mev/sec one day after shutdown.

R_o = 3.5 ft

v = 668.4 ft^3

C = 5.57×10^8 mev/ft^2-sec/r/hr (Reference 32).

μ_c = 4.69 (ft^{-1}) (Reference 33).

μ_s = 2.15 (ft^{-1}) (Reference 33).

The remaining input values are tabulated below.

β	a (ft)	Z (ft)	t_c (ft)	(θ)	$F[\theta(\mu_c t_c + \mu_s Z)]$ (s)
3.88	1	1.601	1	83.3	1.4×10^{-4}
7.89	2	1.526	2	76.7	1.1×10^{-6}
13.65	3	1.507	3	70.55	8.7×10^{-9}
22.01	44	1.552	4	64.8	6.1×10^{-11}

The results of the analysis of the shielding requirements for the 5000 gal waste tank are shown in Fig. 14-32. It can be seen that the 5000 gal tank, under the assumed conditions, will require 2.3 ft of concrete shielding or its equivalent.

The values of the input data for the calculation of the shielding requirements for the 2500 gal waste tank are tabulated below. Factors which retain the same values as listed in the preceding discussion are omitted.

R (ft)	V (ft^3)	Z (ft)	θ	$F[\theta(\mu_c t_c + \mu_s Z)]$
3.0	334.2	1.516	79.7	1.4×10^{-4}
3.0	334.2	1.421	70.0	1.3×10^{-6}
3.0	334.2	1.390	61.35	1.1×10^{-8}
3.0	334.2	1.414	53.8	8.3×10^{-11}

The results of the 2500 gal tank calculation are shown in Fig. 14-33. It can be seen that the 2500 gal waste tank will require 2.4 ft of concrete shielding or its equivalent. This same shielding is sufficient to reduce the dose rate due to 20 percent of the fission products from one fuel plate to 3 mr/hr.

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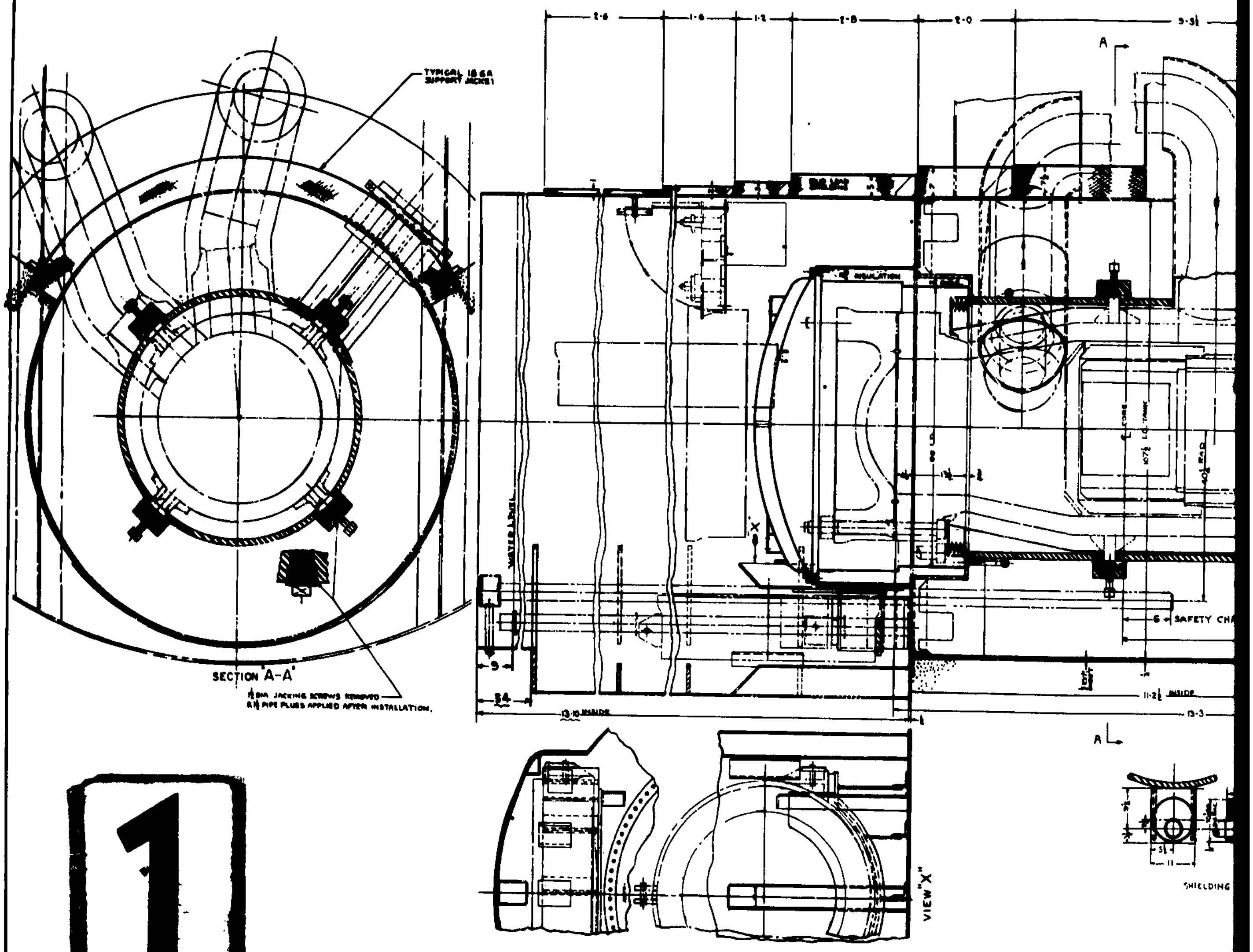
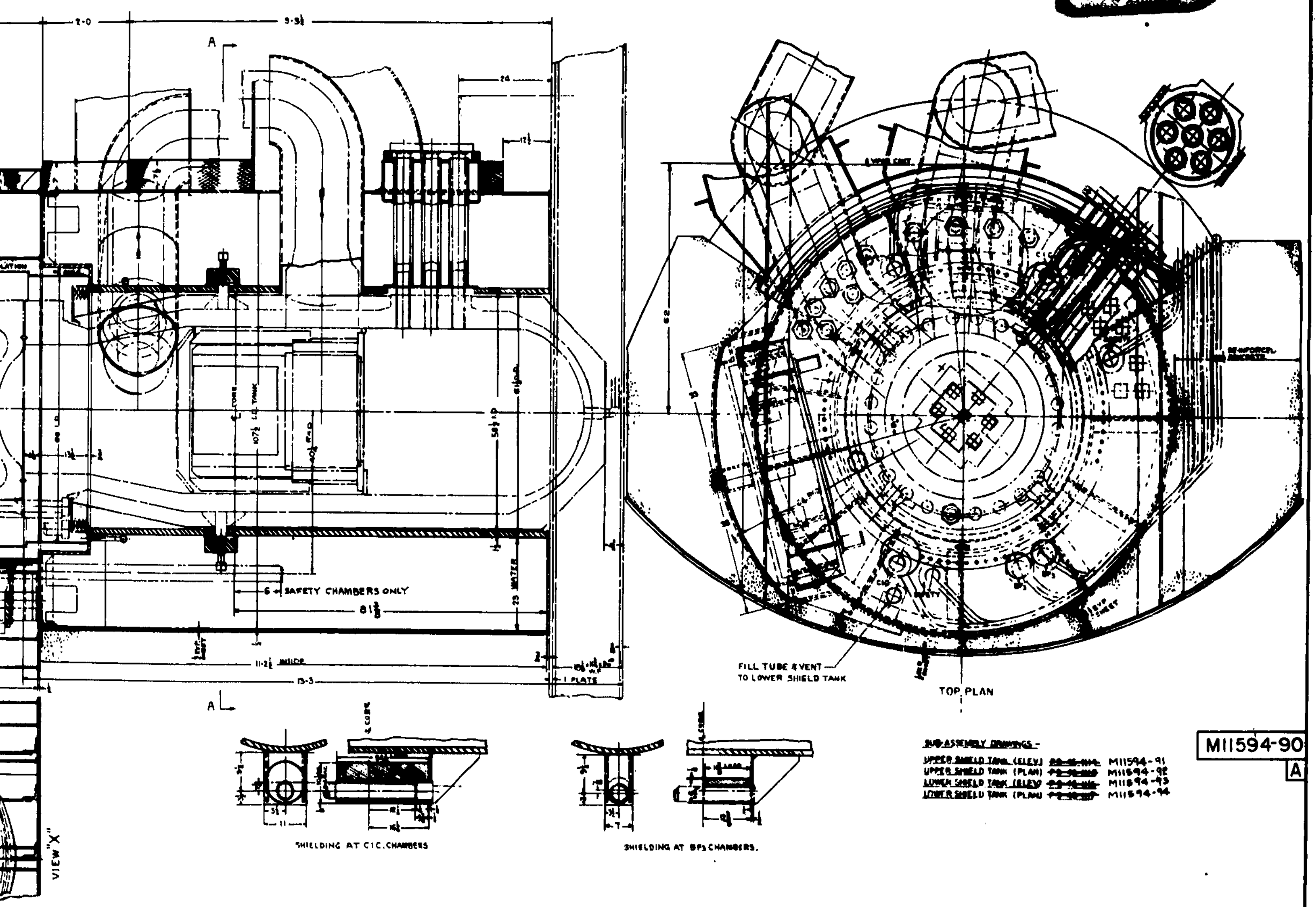

TYPICAL 18 6A SUPPORT JACKS
SECTION 'A-A'
INSULATION
WATER LEVEL
SAFETY CH
13-10 INSIDE
11-2½ INSIDE
13-3
VIEW 'X'
SHIELDING

1

A	ADDED DES. NO. M11594-90



SUB-ASSEMBLY DRAWINGS -

UPPER SHIELD TANK (ELEV) M11594-91
UPPER SHIELD TANK (PLAN) M11594-92
LOWER SHIELD TANK (ELEV) M11594-93
LOWER SHIELD TANK (PLAN) M11594-94

M11594-90
A

LIST OF MATERIAL

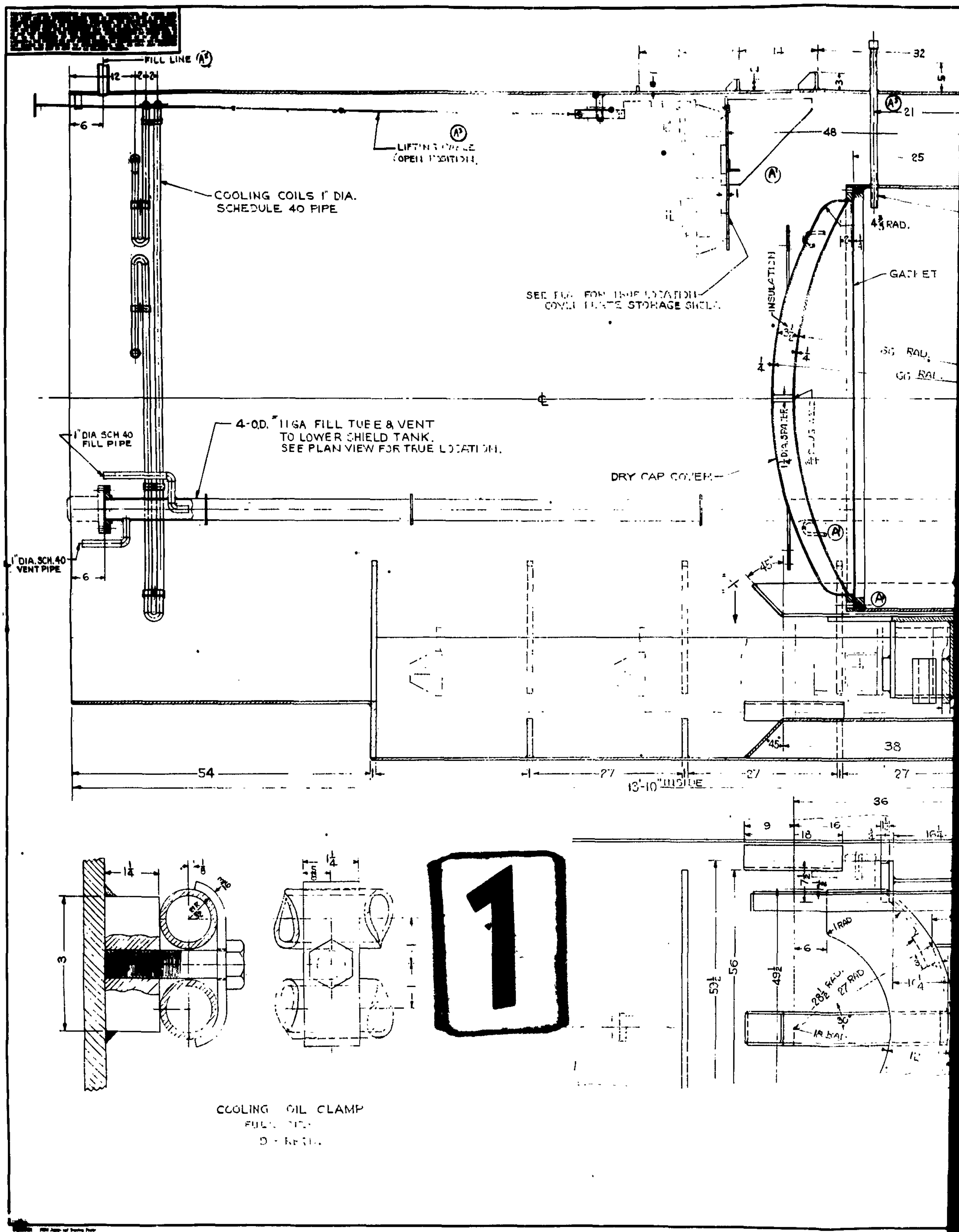
APPLICATION

QTY REQD

PRIMARY SHIELDING ARRANGEMENT

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-90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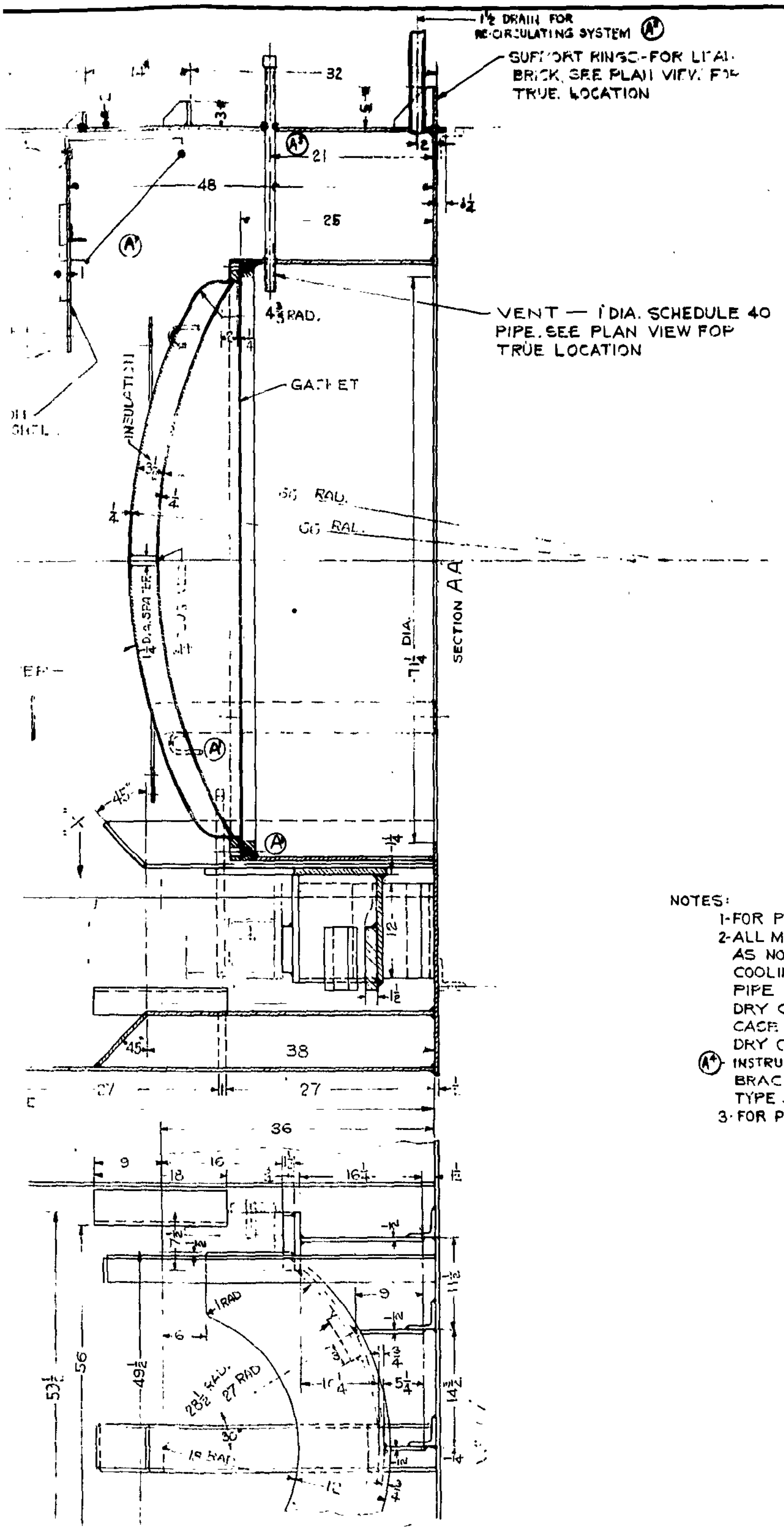
FILL LINE
COOLING COILS 1" DIA.
SCHEDULE 40 PIPE
STORAGE SHIELD
INSULATION
GASKET
4$\frac{3}{4}$ RAD.
4-O.D. 11 GA FILL TUBE & VENT
TO LOWER SHIELD TANK.
SEE PLAN VIEW FOR TRUE LOCATION.
1" DIA SCH 40
FILL PIPE
1" DIA. SCH. 40
VENT PIPE
DRY CAP COVER
13'-10" INSIDE
1
COOLING
CLAMP

A	ON DRY CAP COVER		
	COVER PLATE SHELF BRACKET CHGD.		
	COVER BOLT CIRCLE CHGD. (A)		
	1½ RE-CIRCULATING LINE ADDED (A)		
	LIFTING CABLE ETC. ADDED (A)		
	REMOVED MATL. SPEC. ON ENT.		
	REPLACING WITH INST. TUBE SPEC. (A)		
B	ADDED DRG NO M11594-91 ECO 0298		



NOTES:

1-FOR PLAN VIEW SEE DRG M11594-92

2-ALL MATERIAL TO BE ASTM A240, TYPE 304 EXCEPT AS NOTED FOR THE FOLLOWING PARTS:
COOLING COILS-ASTM A 312, TYPE 304.
PIPE FITTINGS-ASTM A 182, F 304.
DRY CAP COVER BOLTS-AISI TYPE 403 TO ASME CASE 1274 N.
DRY CAP GASKET-STAINLESS STL. TYPE 304 METAL-ASBESTOS.
(A) INSTRUMENT TUBE WELLS & FILL TUBE · ASTM A-269 TYPE 304.
BRACKETS, STORAGE SHELVES, INTERIOR GUSSETS-ASTM A240, TYPE 304 OR ASTM A 167, GRADE 3.

3-FOR PRIMARY SHIELDING ARRGT. SEE DRG M11594-90

M11594-91
B

LIST OF MATERIAL

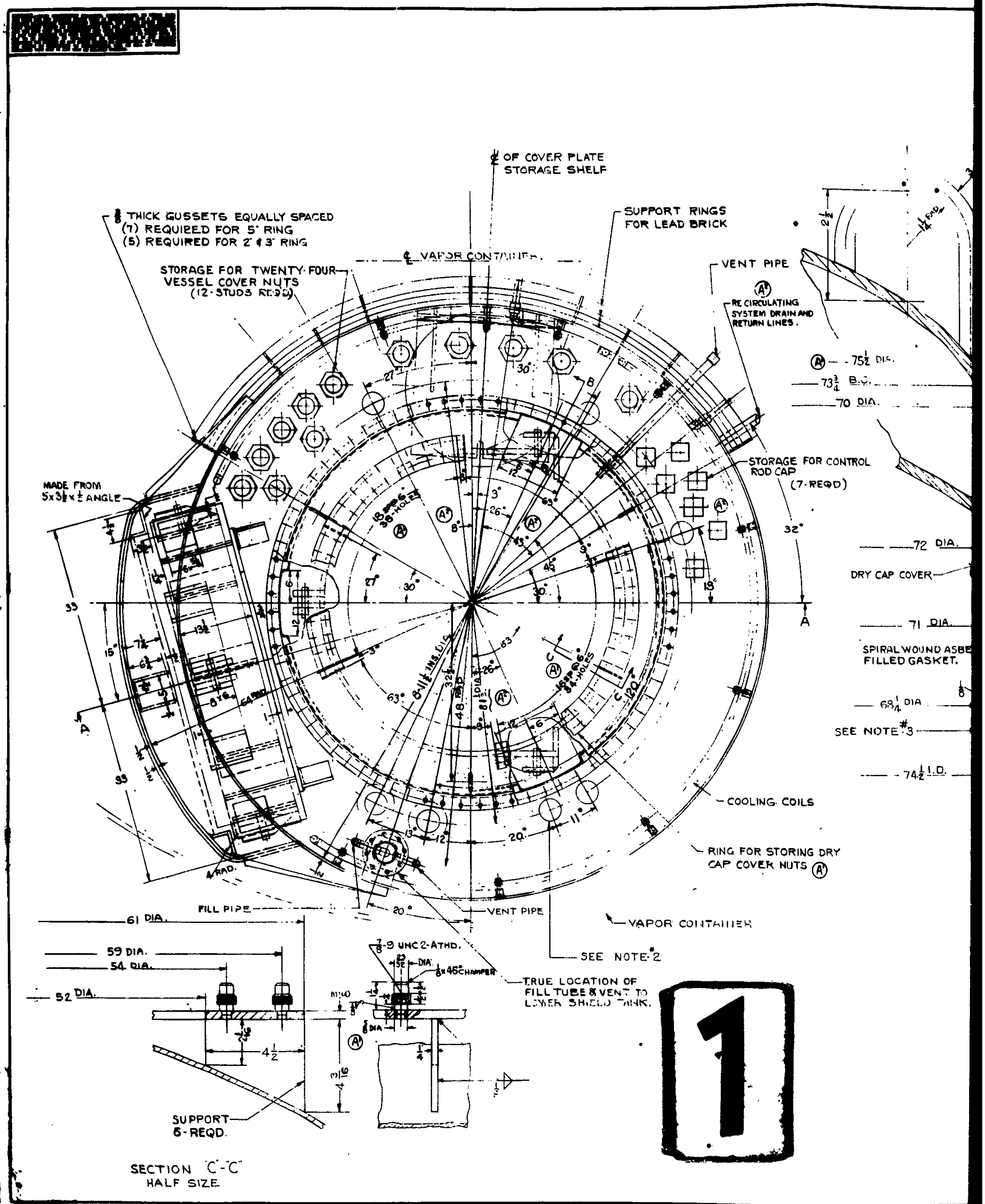
UPPER SHIELD TANK
ELEVATION

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA.

M11594-91

APPLICATION

QTY REQD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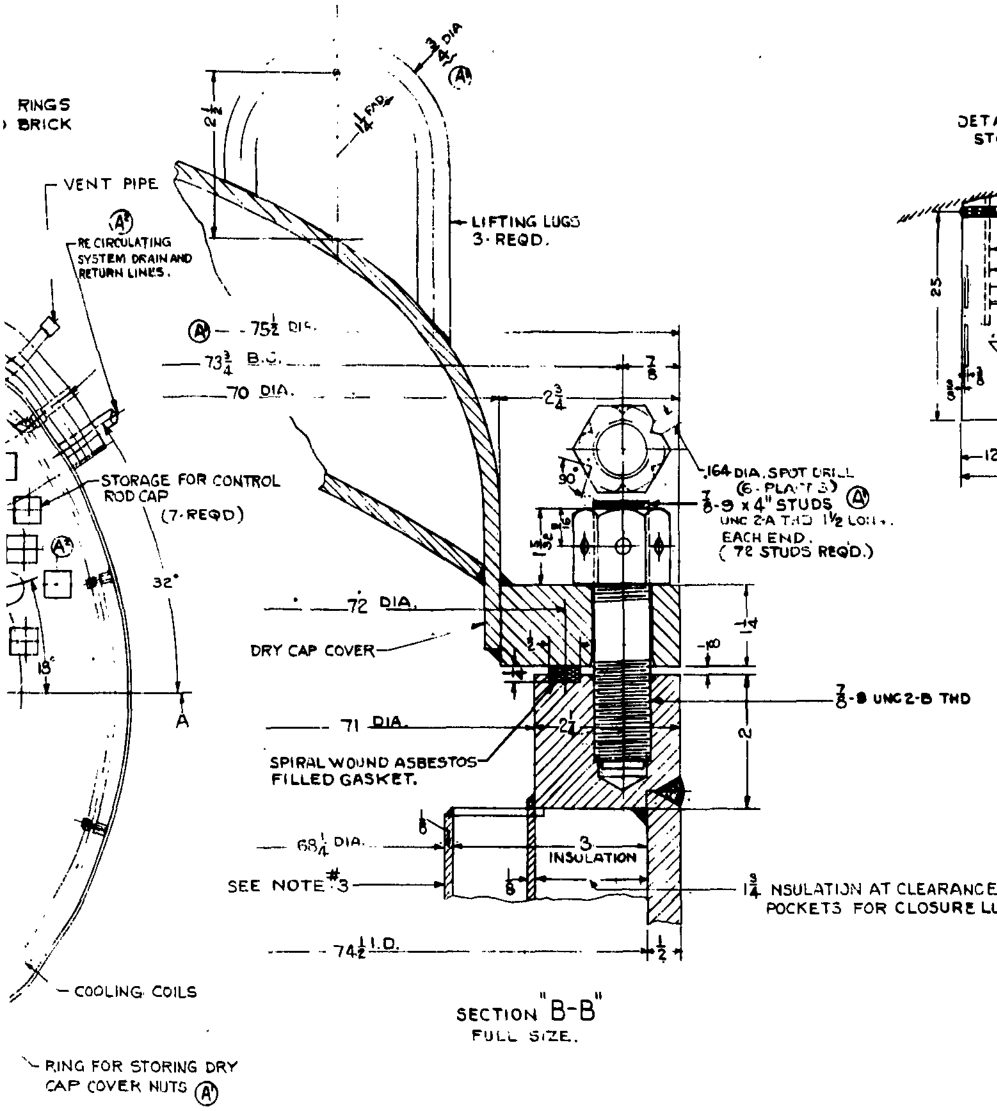
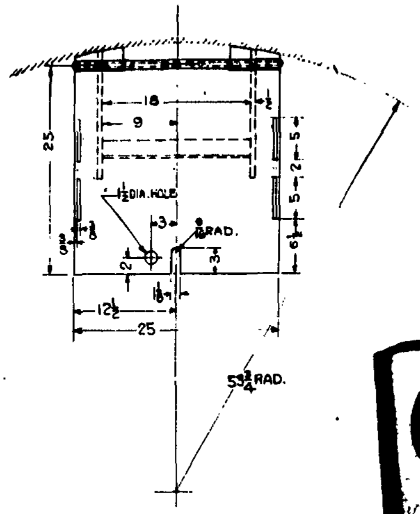
℄ OF COVER PLATE STORAGE SHELF
SUPPORT RINGS FOR LEAD BRICK
3/8 THICK GUSSETS EQUALLY SPACED (7) REQUIRED FOR 5" RING (5) REQUIRED FOR 2" & 3" RING
STORAGE FOR TWENTY-FOUR VESSEL COVER NUTS (12-STUDS REQD)
℄ VAPOR CONTAINER
VENT PIPE
RE CIRCULATING SYSTEM DRAIN AND RETURN LINES.
75 1/2 DIA.
73 3/4 B.C.
70 DIA.
STORAGE FOR CONTROL ROD CAP (7-REQD)
MADE FROM 5x3 1/2 x 1/2 ANGLE
72 DIA.
DRY CAP COVER
71 DIA.
SPIRAL WOUND ASBE FILLED GASKET.
68 1/4 DIA
SEE NOTE #3
74 1/2 I.D.
COOLING COILS
RING FOR STORING DRY CAP COVER NUTS
FILL PIPE
VENT PIPE
VAPOR CONTAINER
SEE NOTE #2
61 DIA.
59 DIA.
54 DIA.
52 DIA.
7/8-9 UNC 2-A THD.
TRUE LOCATION OF FILL TUBE & VENT TO LOWER SHIELD TANK.
SUPPORT 6-REQD.
SECTION "C-C" HALF SIZE
1

A	CHGD ORIENTATION ...
	LIFTING LUGS ... DRY CAP COVER ...
	... NUT ... 9 STUDS WERE ... 10 BOLTS
	RE-DESIGNED COVER & GASKET
	B&45 ADDED, RECIRC. LINES ADDED.
	CONT. ROD STORAGE CHGD. ...
	NOTE #2 CHGD. NOTE #5 ADDED (A)
B	ADDED DRG. NO M11594-92 ...



SECTION "B-B"
FULL SIZE.

DETAIL OF COVER PLATE
STORAGE SHELF



M11594-92
B

NOTES-
#1. FOR ELEVATION SEE DRG. M11594-91
#2. HOLES FOR 4 IN. O.D. INSTRUMENT TUBE WELLS (7-REQD.) (A)
#3 INSULATION RETAINER SHOWN FOR FABRICATION PURPOSES ONLY RETAINER SHALL BE SHIPPED SEPARATELY AND INSTALLED IN FIELD.
#4 FOR PRIMARY SHIELDING ARRGT. SEE DRG M11594-90
(A) #5 FOR ARRGT. OF FUEL TRANSFER SYSTEM SEE DRG M11594-64

LIST OF MATERIAL

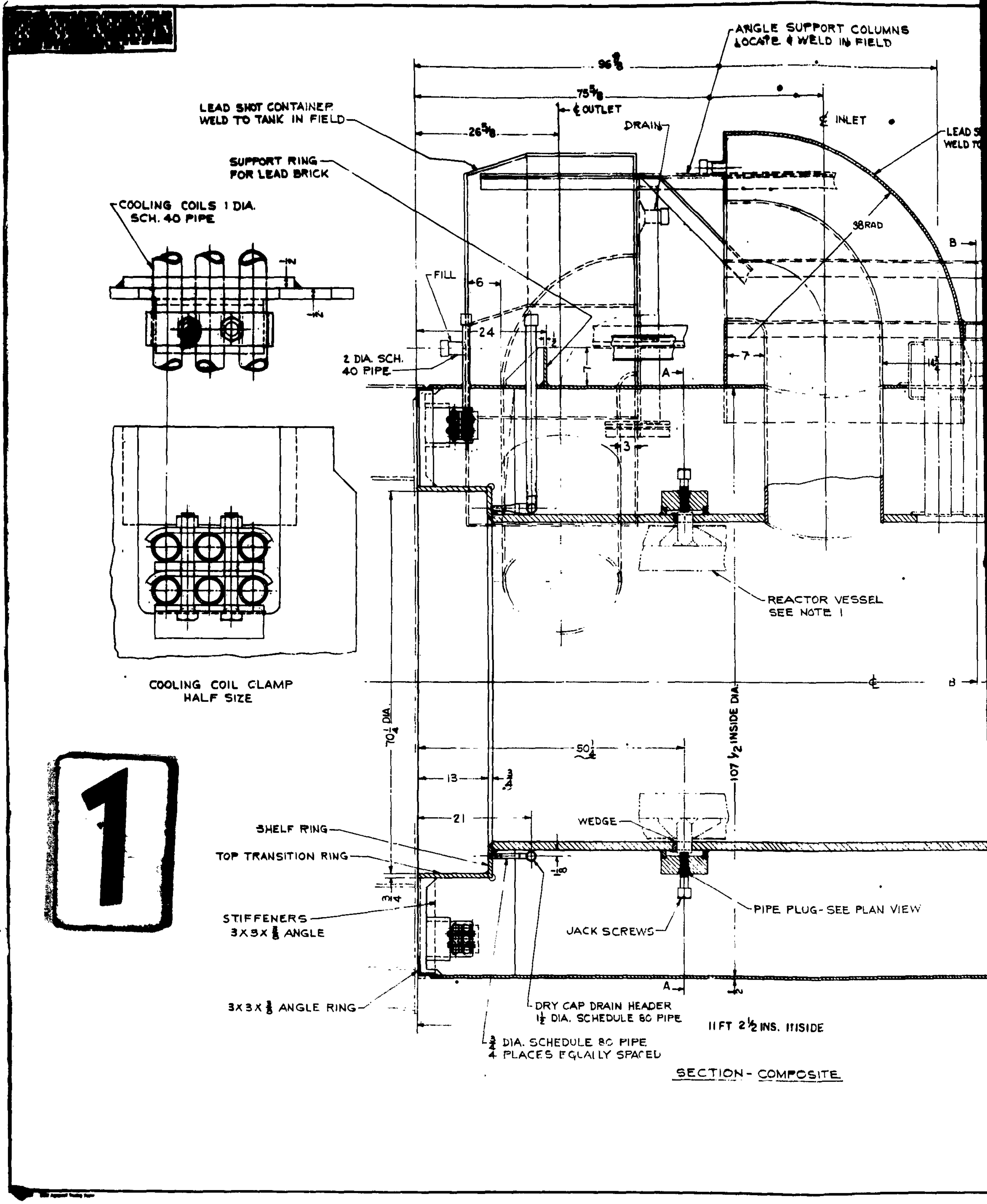
APPLICATION | QTY REQD

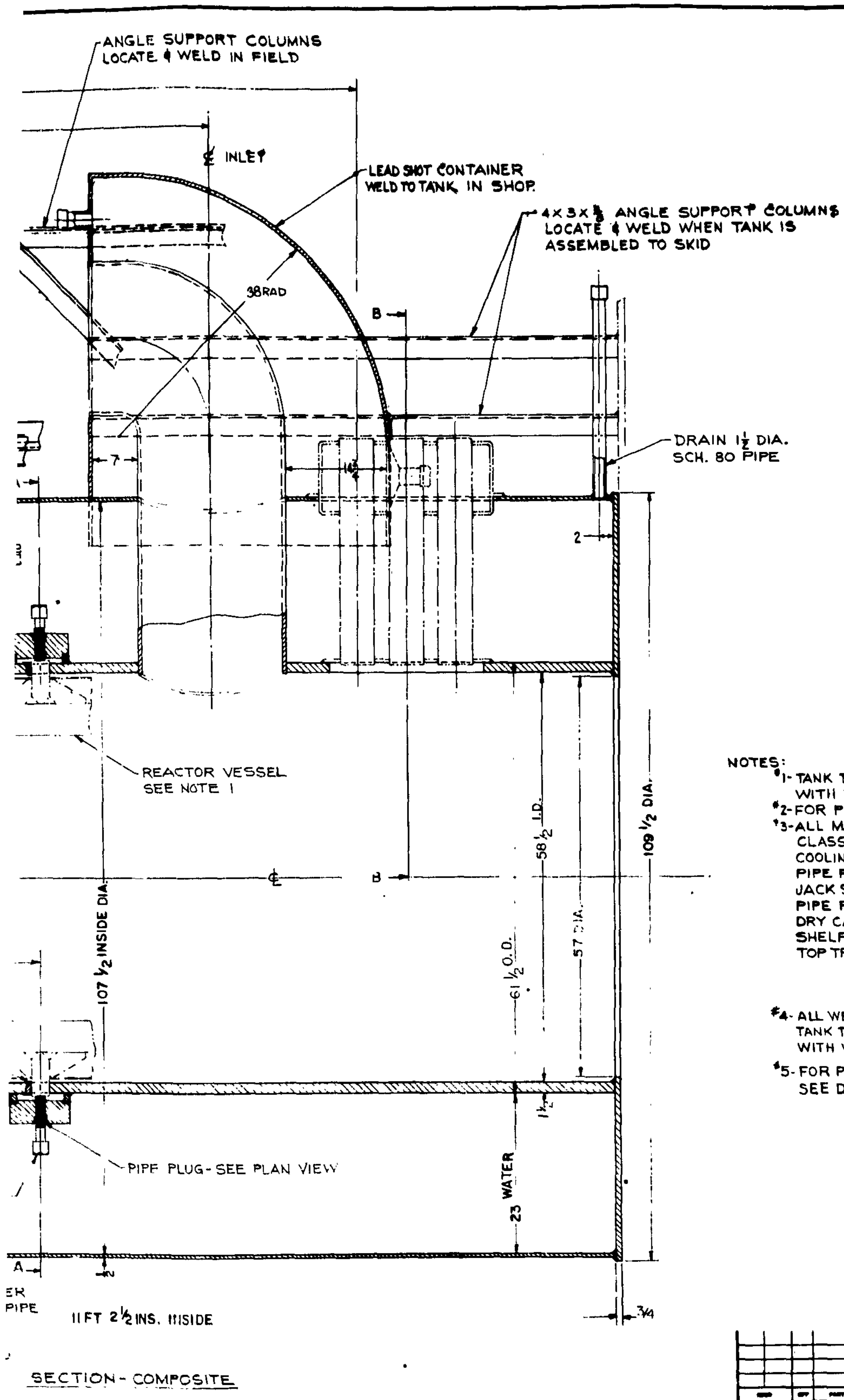
UPPER SHIELD TANK
PLAN

U. S. ARMY
CORPS OF ENGINEERS
FORT BELVOIR, VA.

B M11594-92

SCALE 1½" = 1'-0"


ANGLE SUPPORT COLUMNS
LOCATE & WELD IN FIELD
LEAD SHOT CONTAINER
WELD TO TANK IN FIELD
℄ OUTLET
DRAIN
℄ INLET
SUPPORT RING
FOR LEAD BRICK
COOLING COILS 1 DIA.
SCH. 40 PIPE
38 RAD
FILL
2 DIA. SCH.
40 PIPE
REACTOR VESSEL
SEE NOTE 1
COOLING COIL CLAMP
HALF SIZE
$70\frac{1}{4}$ DIA.
$107\frac{1}{2}$ INSIDE DIA.
1
SHELF RING
TOP TRANSITION RING
WEDGE
STIFFENERS
3 X 3 X $\frac{3}{8}$ ANGLE
JACK SCREWS
PIPE PLUG - SEE PLAN VIEW
3 X 3 X $\frac{3}{8}$ ANGLE RING
DRY CAP DRAIN HEADER
$1\frac{1}{2}$ DIA. SCHEDULE 80 PIPE
11 FT $2\frac{1}{2}$ INS. INSIDE
$\frac{3}{4}$ DIA. SCHEDULE 80 PIPE
4 PLACES EQUALLY SPACED
SECTION - COMPOSITE



SECTION - COMPOSITE

LTR	REVISION
A	ADDED DRG. NO. M11594-93

NOTES:

*1- TANK TO BE FABRICATED AROUND REACTOR VESSEL WITH VESSEL IN PLACE AS SHOWN.

*2- FOR PLAN VIEW, SECT. A-A & SECT. B-B SEE DRG. M11594-94

*3- ALL MATERIAL TO BE ASTM A201, GRADE B TO A-300 CLASS 1, EXCEPT AS NOTED FOR THE FOLLOWING PARTS:
COOLING COILS & DRAIN - ASTM 106, GRADE B.
PIPE FITTINGS - ASTM A 105, GRADE II.
JACK SCREWS - ASTM A 107, GRADE 1020.
PIPE PLUGS - ASTM A 338-54.
DRY CAP DRAIN HEADER - ASTM A 312, TYPE 304.
SHELF RING - ASTM A 240, TYPE 304.
TOP TRANSITION RING - ASTM A 240, TYPE 304.

*4- ALL WELDS TO BE WATER TIGHT. TANK TO BE LEAK TESTED BY FILLING WITH WATER TO OPERATING LEVEL.

*5- FOR PRIMARY SHIELDING ARRGT. SEE DRG M11594-90

M11594-93

LIST OF MATERIAL

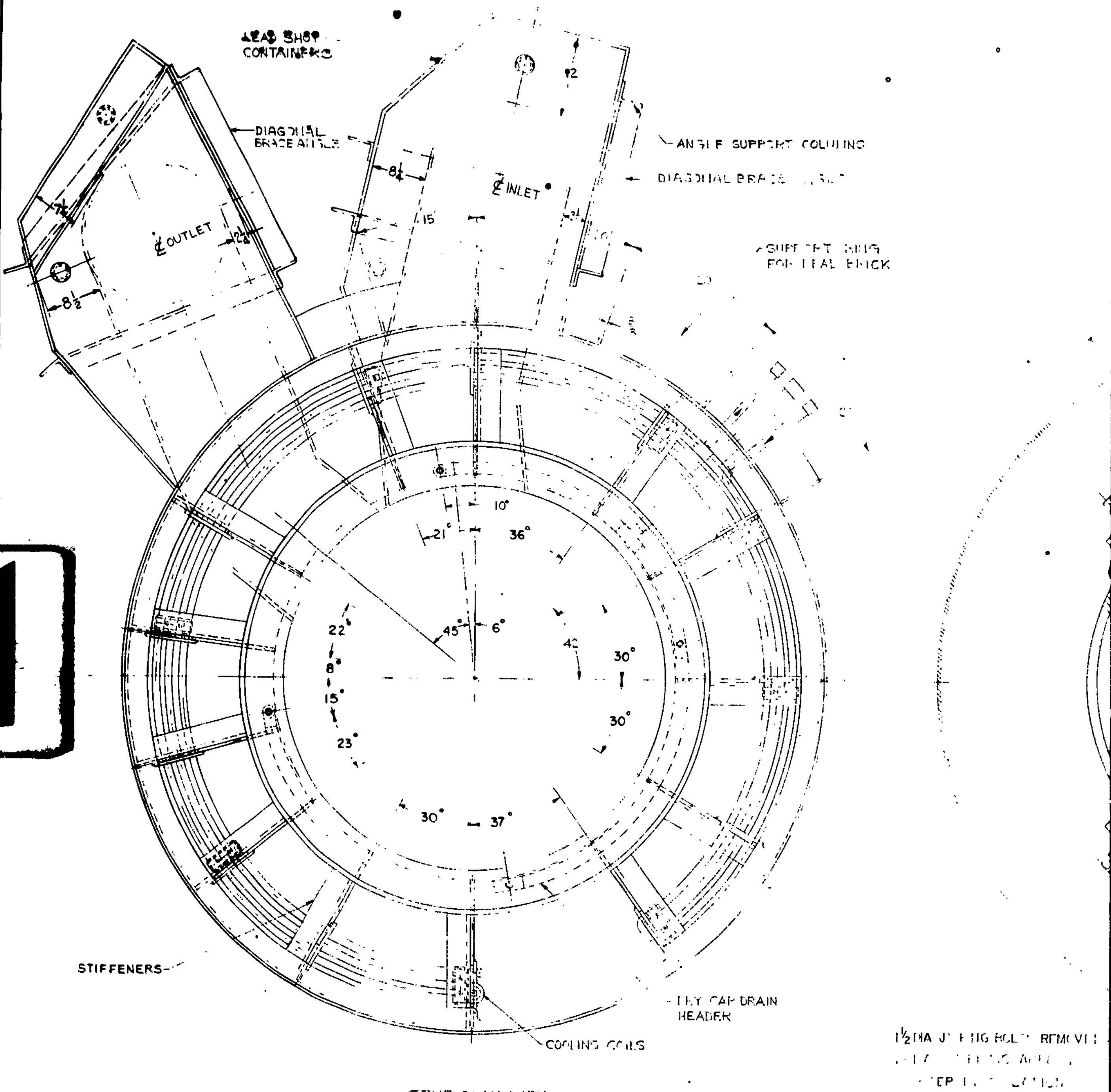
APPLICATION | QTY REQD

LOWER SHIELD TANK
ELEVATION

U.S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.

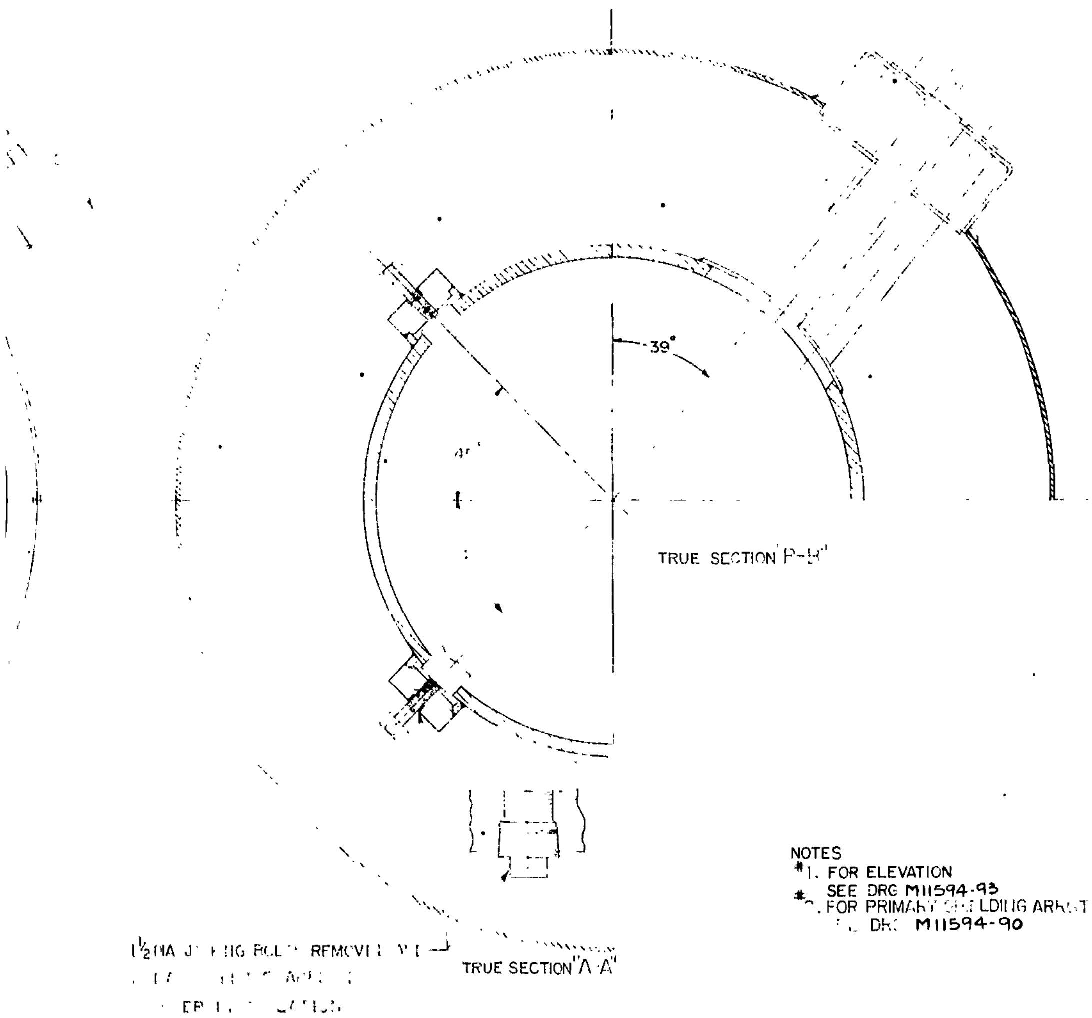
SCALE 1½" = 1'-0"

M11594-93



TRUE PLAN VIEW

REV	DESCRIPTION	DATE	APPROVAL	
A	ADDED DRG. NO. M11594-94			



2

M11594-94
A

NOTES
*1. FOR ELEVATION SEE DRG M11594-93
*2. FOR PRIMARY SHIELDING ARR'T SEE DRG M11594-90

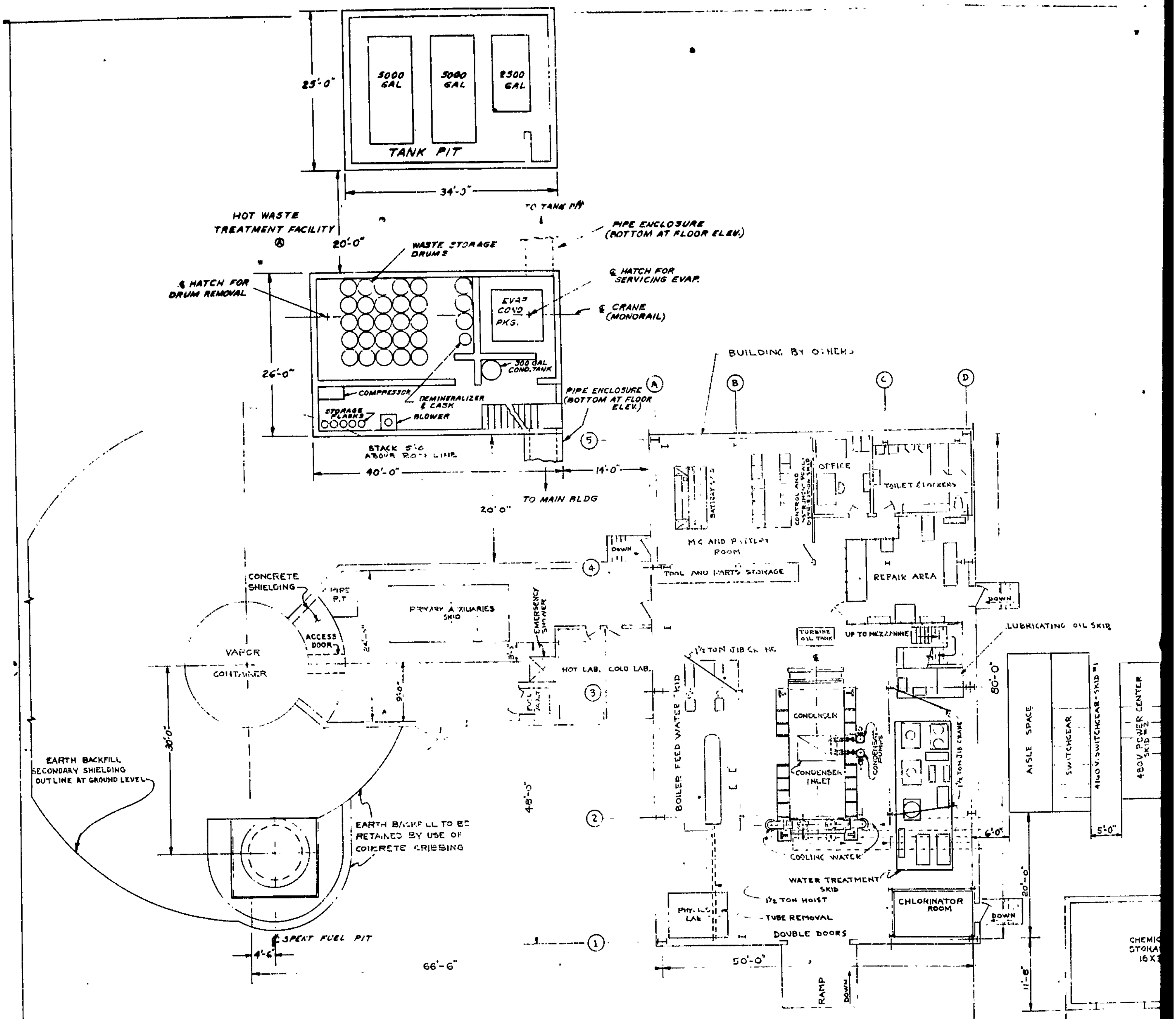
LIST OF MATERIAL

QTY	PART NUMBER	NEXT ASSY	USED ON	NEXT ASSY	FINAL ASSY
		APPLICATION		QTY REQD	

LOWER SHIELD TANK
PLAN

U. S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS

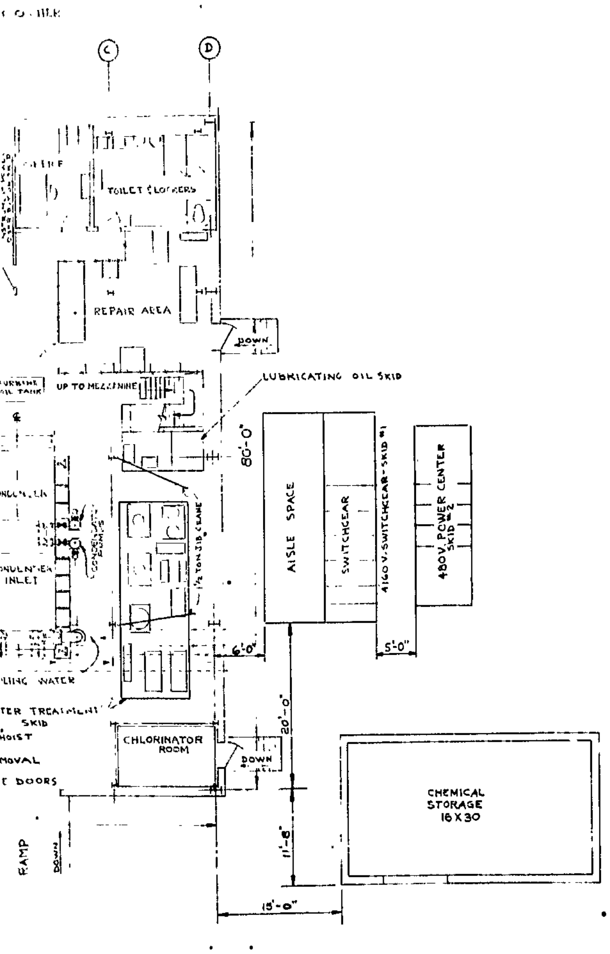
M11594-94

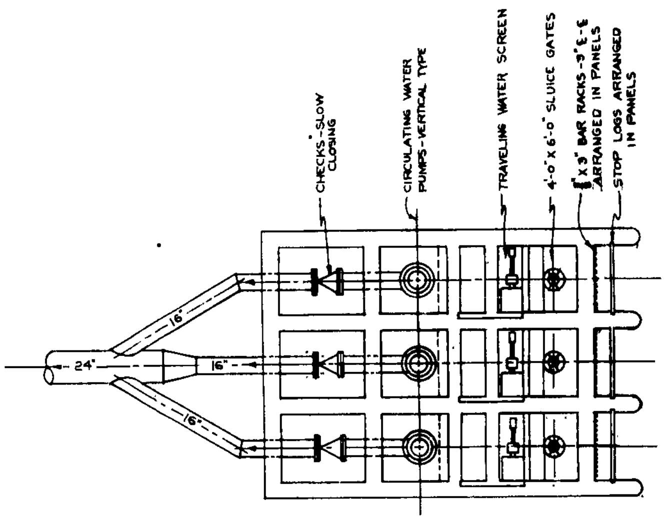


STATION DEVELOPMENT PLAN
SHOWING
LOCATIONS OF ALL PRINCIPAL FEATURES
AS DETERMINED FROM
ALL PRELIMINARY LAYOUT STUDIES
AS ILLUSTRATED BY DRAWINGS MI504-20, 21, 22

1

A	RELOCATED [illegible] DELETED [illegible] ADDED [illegible] FACILITY	WFS	
B	CHANGED SIZE OF WASTE DISPOSAL BLDG. & ARRANGEMENT OF SAME		
C	ADDED DRG. NO M 11594-6 ℄ OF SPENT FUEL TANK ADDED	ECO 0292	





PLAN AT DECK LEVEL
CIRCULATING WATER
INTAKE STRUCTURE
SUPERSTRUCTURE ENCLOSURE
TRASH RACK RAKE AND MONORAIL HOISTS NOT SHOWN

NOTE:

1. THIS DRAWING REPRESENTS OVERALL PLANT LAYOUT & GRADE BASED ON UTILIZATION OF A SINGLE NUCLEAR REACTOR DEVELOPING 28 MW AND A SINGLE TURBO-GENERATOR DEVELOPING 6000 NET eKW.

THE APPROX. LOCATION OF THE INTAKE STRUCTURE IS LOCATED ON THE SHORELINE APPROX. 500'-0" FROM POWER PLANT.

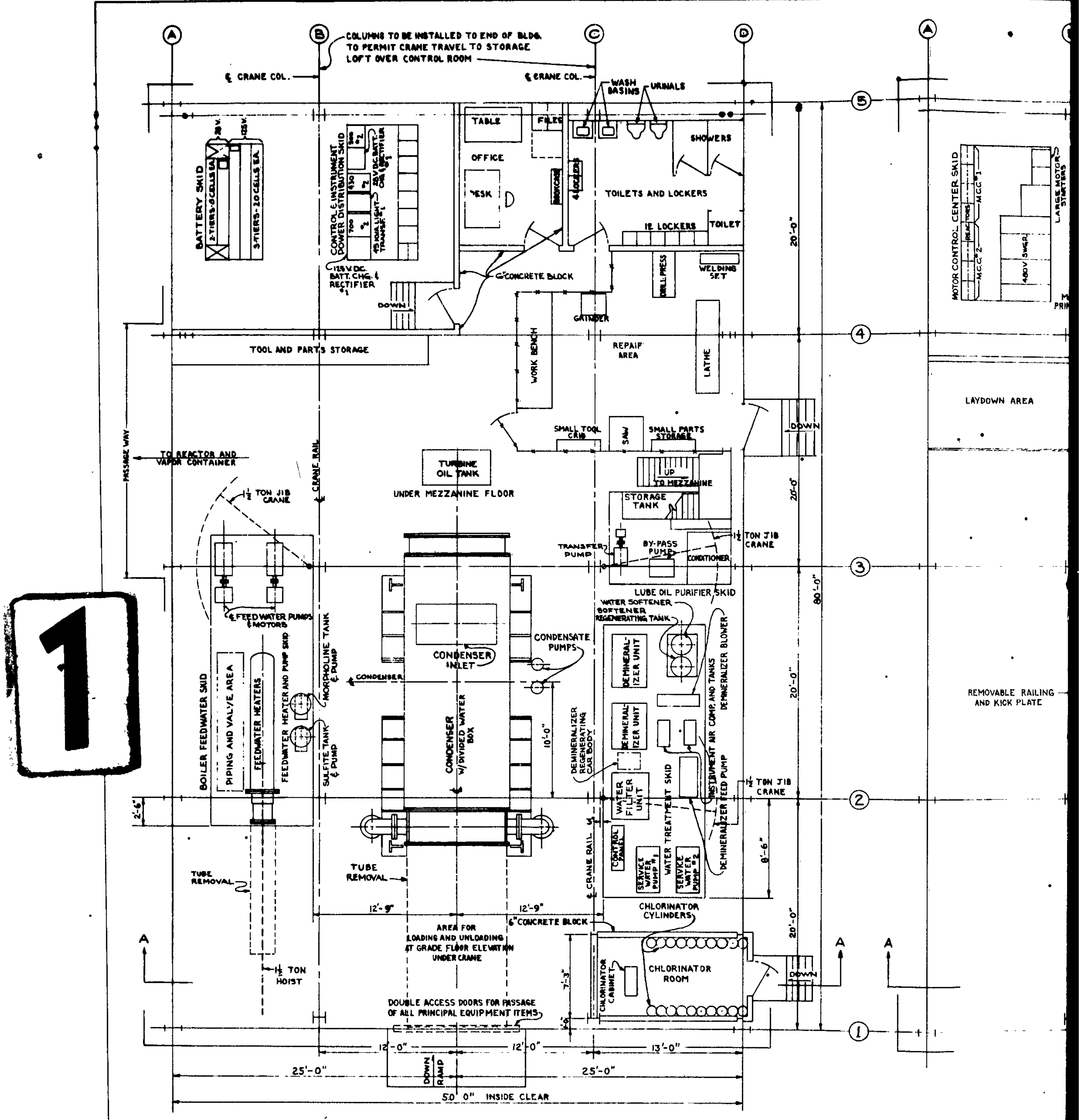
James L. [illegible] 4-21-60
J.N.F.
J.M. [illegible]
R.L. [illegible]
M.O. Hamilak 4-5-60
[illegible] 7/6/60

SM-2 — ONE 7,500 KW. UNIT
DEVELOPMENT PLAN
FOR
EQUIPMENT and FACILITIES
PRIMARY and SECONDARY
INSTALLATIONS

1/8" = 1'-0"

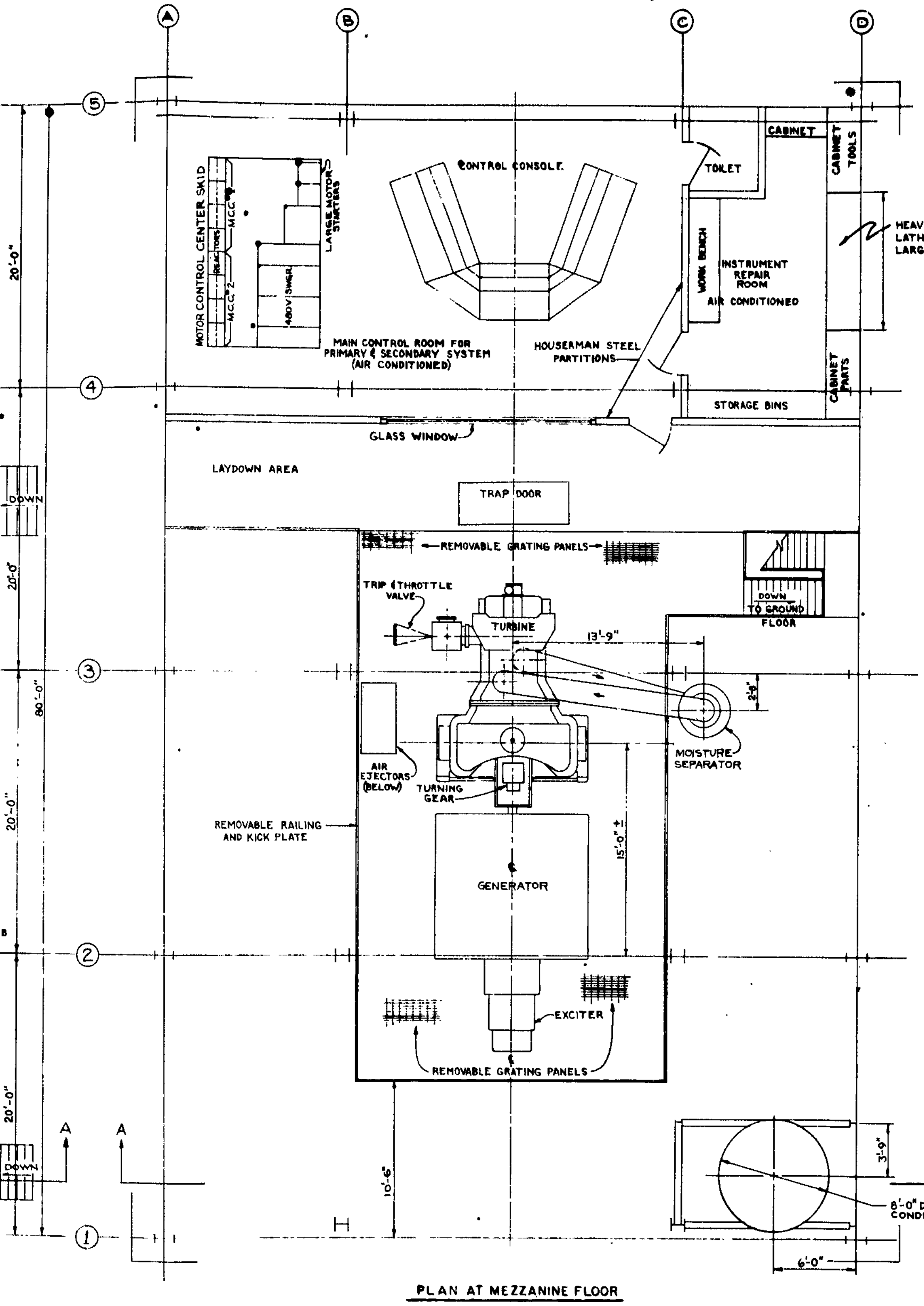
U.S. ARMY
NUCLEAR POWER
FIELD OFFICE
CORPS OF ENGINEERS
FORT BELVOIR, VA

M11594-6



PLAN AT GROUND FLOOR

SEE SHEET M03-M3 FOR SECTION A-A



HEAVY BENCH WITH TOOL ROOM LATHE AND OTHER EQUIPMENT. LARGE WINDOW OVER

PLAN AT MEZZANINE FLOOR

TURBINE GENERATOR DATA

ONE HORIZONTAL TURBINE-GENERATOR, CONDENSING, THREE BEARING TYPE WITH DIRECT CONNECTED EXCITER.

INITIAL PRESSURE, 575 PSIG. AND INITIAL TEMP. 485 DEG.F. 3600 R.P.M. 9,375 K.V.A. AND 7500 K.W. AT .8P.F. THREE PHASE, 60 CYCLE, 4,160 VOLTS. WITH TURNING GEAR. ARRANGED FOR 3 EXTRACTION POINTS FOR FEEDWATER HEATING.

POWER PLANT BUILDING AND FOUNDATIONS WILL BE DESIGNED AND CONSTRUCTED UNDER SEPARATE CONTRACTS.

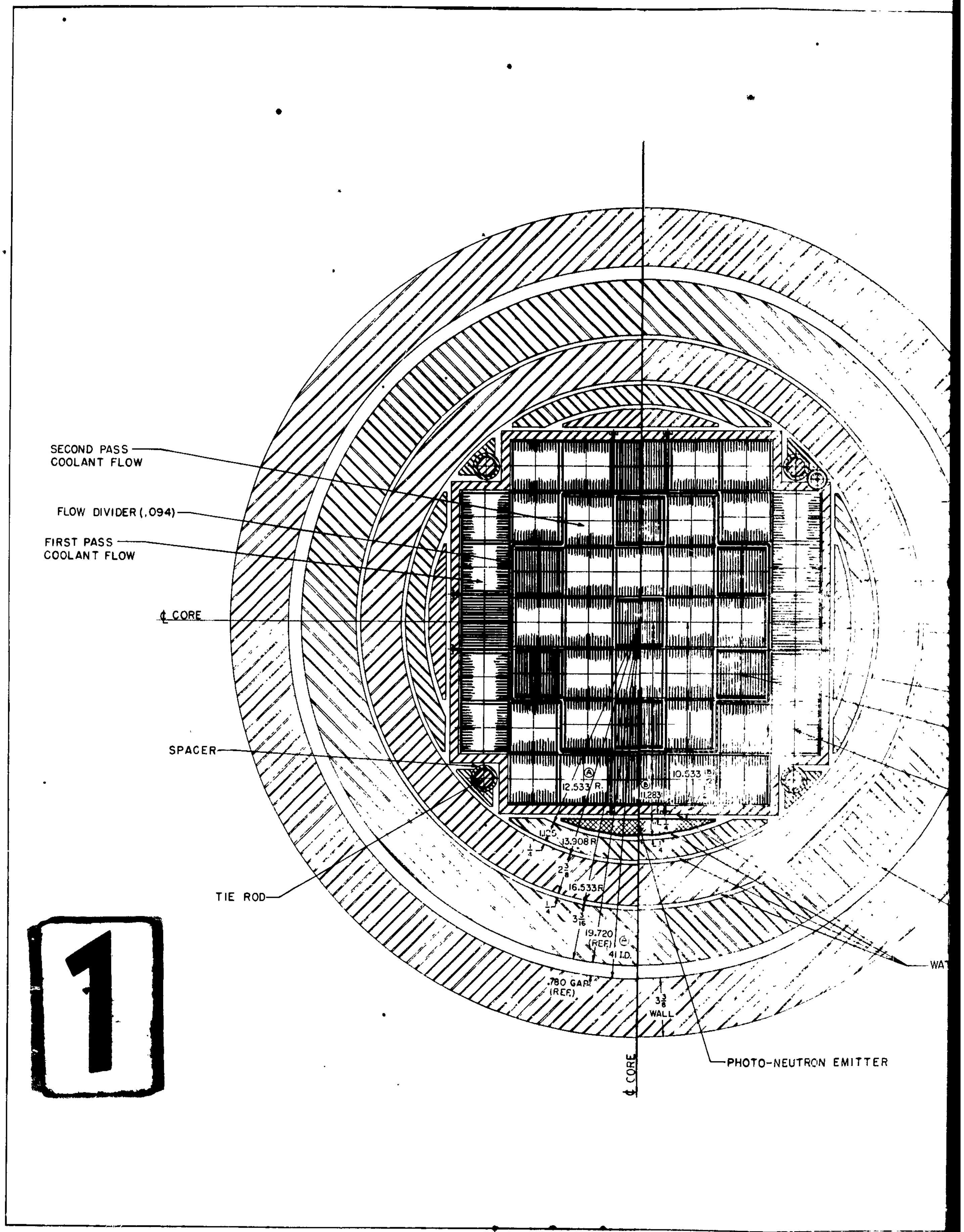
APPLICATION | QTY REQD

SM-2 -- ONE 7,500 K.W. UNIT
GENERAL ARRANGEMENT
PLANS SHOWING LAYOUTS FOR GROUND FLOOR and MEZZANINE SECONDARY INSTALLATIONS CENTRAL CONTROL CONSOLE

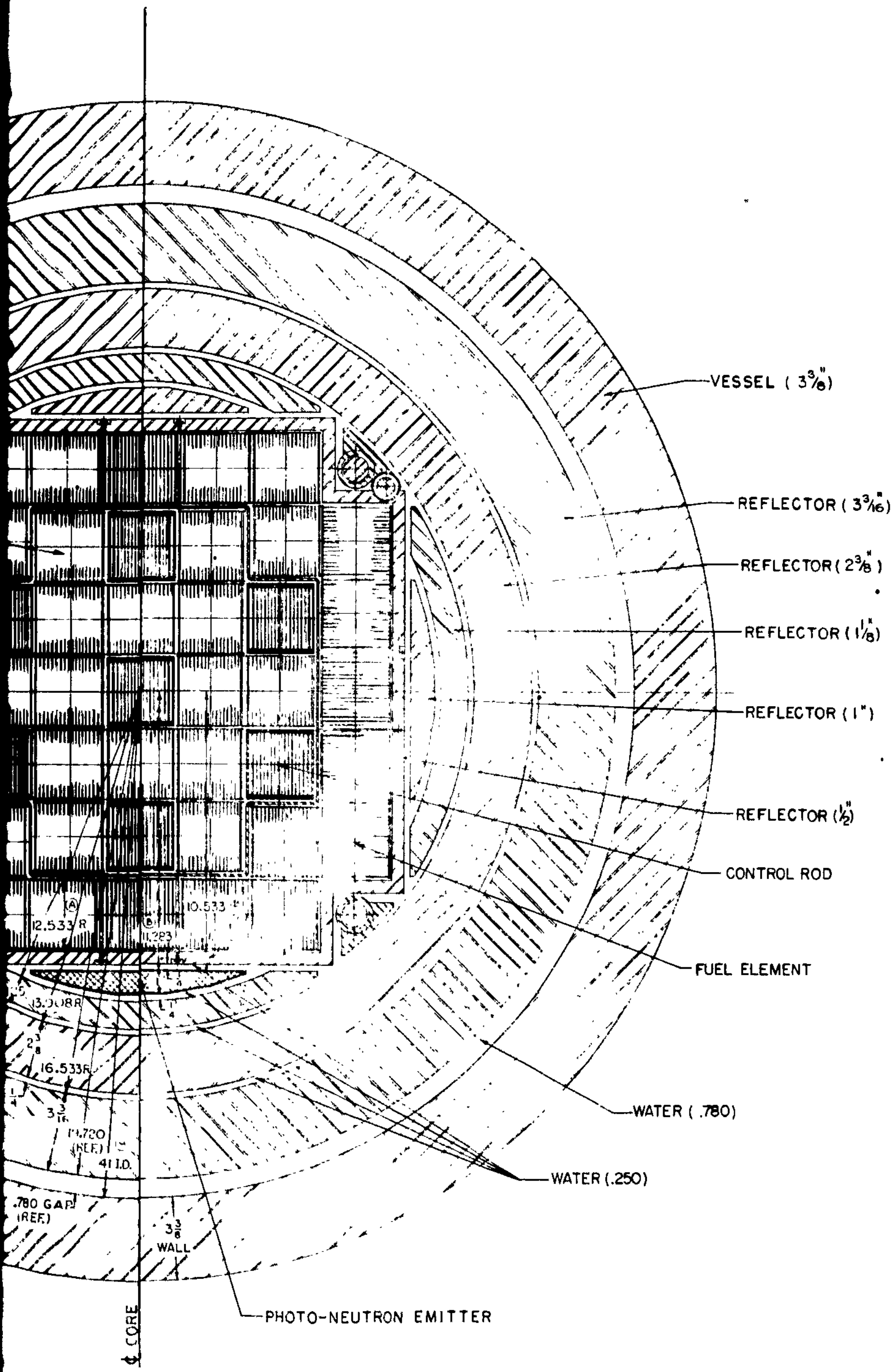
SCALE 1/4" = 1'-0"

U. S. ARMY NUCLEAR POWER FIELD OFFICE CORPS OF ENGINEERS FORT BELVOIR, VA.

M 11594-7



SECOND PASS COOLANT FLOW
FLOW DIVIDER (.094)
FIRST PASS COOLANT FLOW
℄ CORE
SPACER
TIE ROD
12.533 R.
10.533
1.283
13.908 R
16.533 R
19.720 (REF.)
41 I.D.
.780 GAP (REF.)
WALL
PHOTO-NEUTRON EMITTER
℄ CORE
1



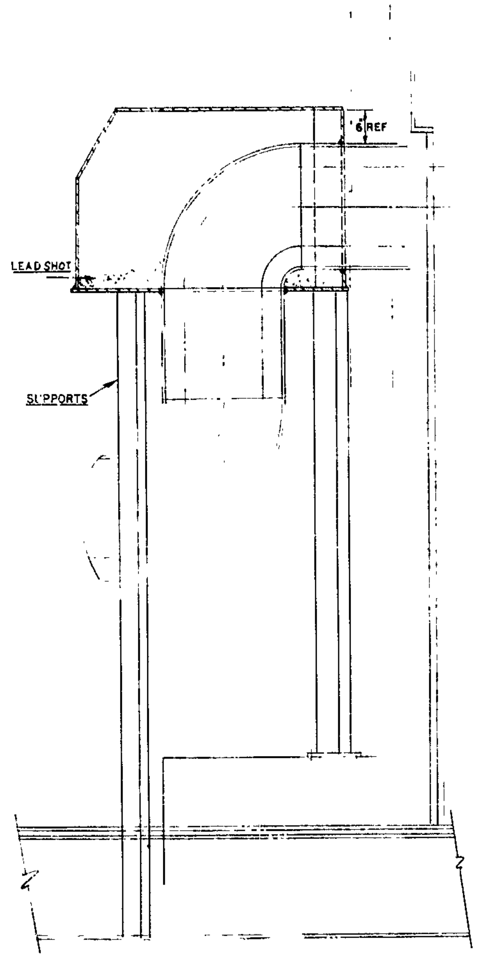
LAYOUT

MODEL SM-2

SCALE HALF

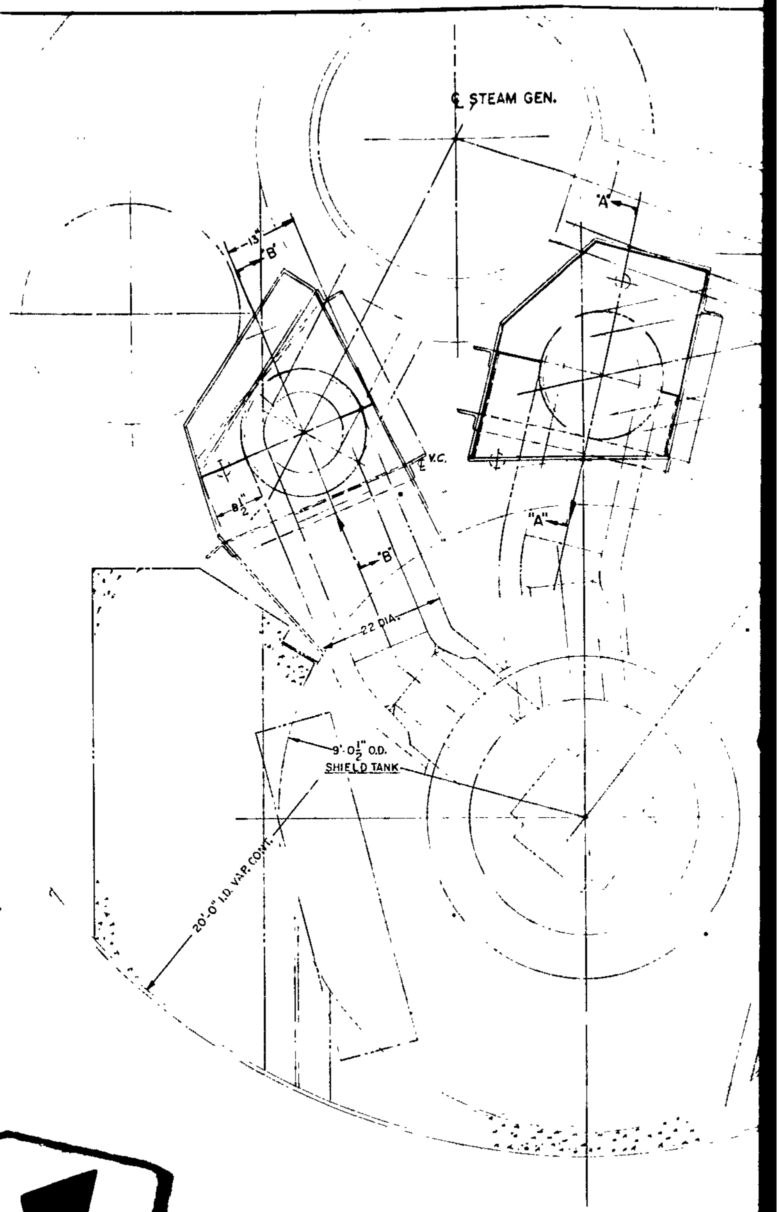
TITLE: REACTOR CROSS SECTION

LAYOUT NO.: AES-341

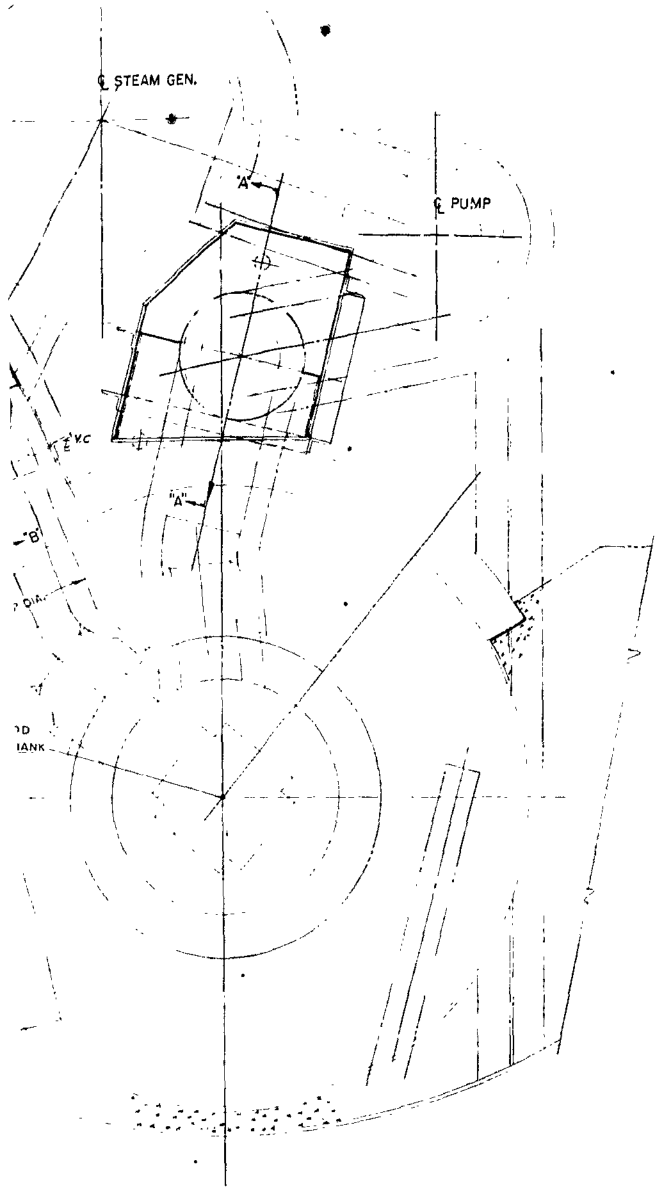


SECTION "B-B"

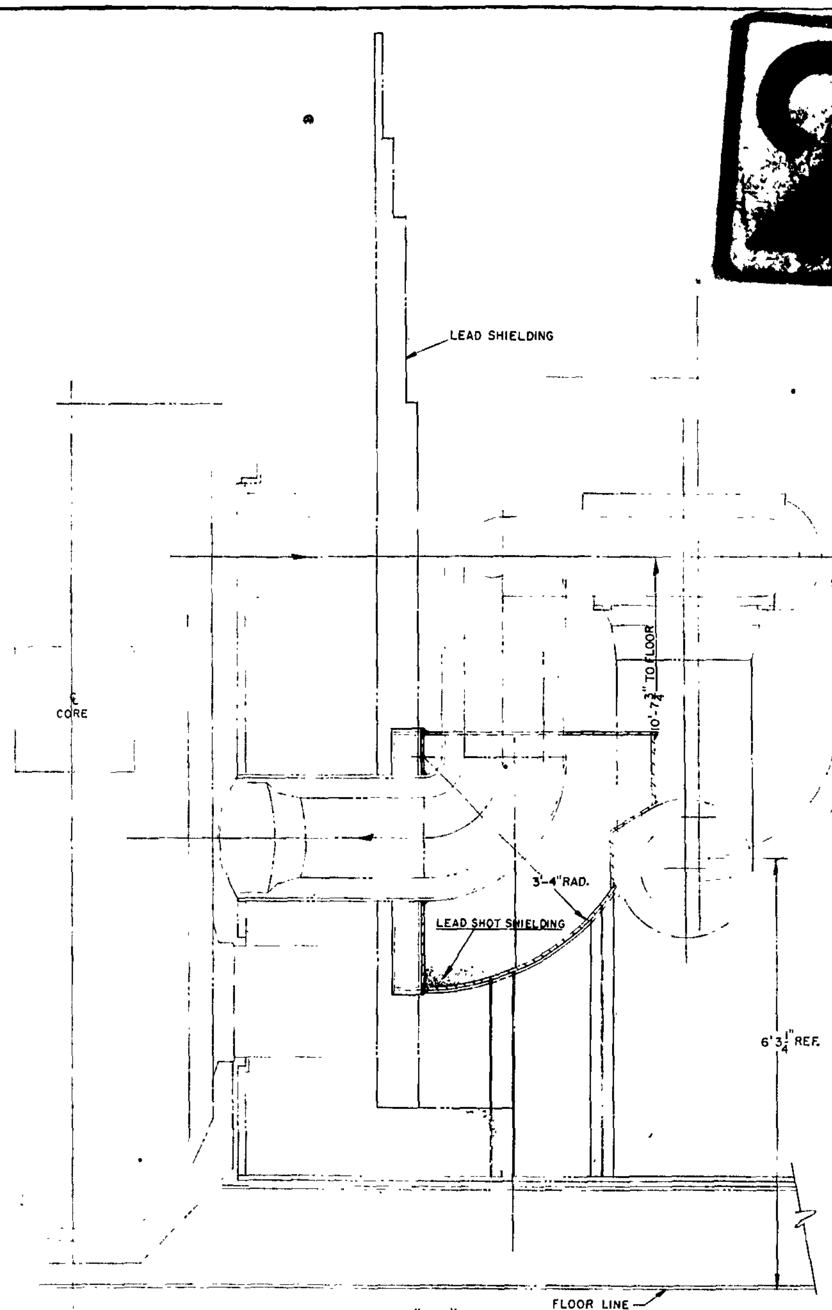
1



PLAN VIEW



PLAN VIEW

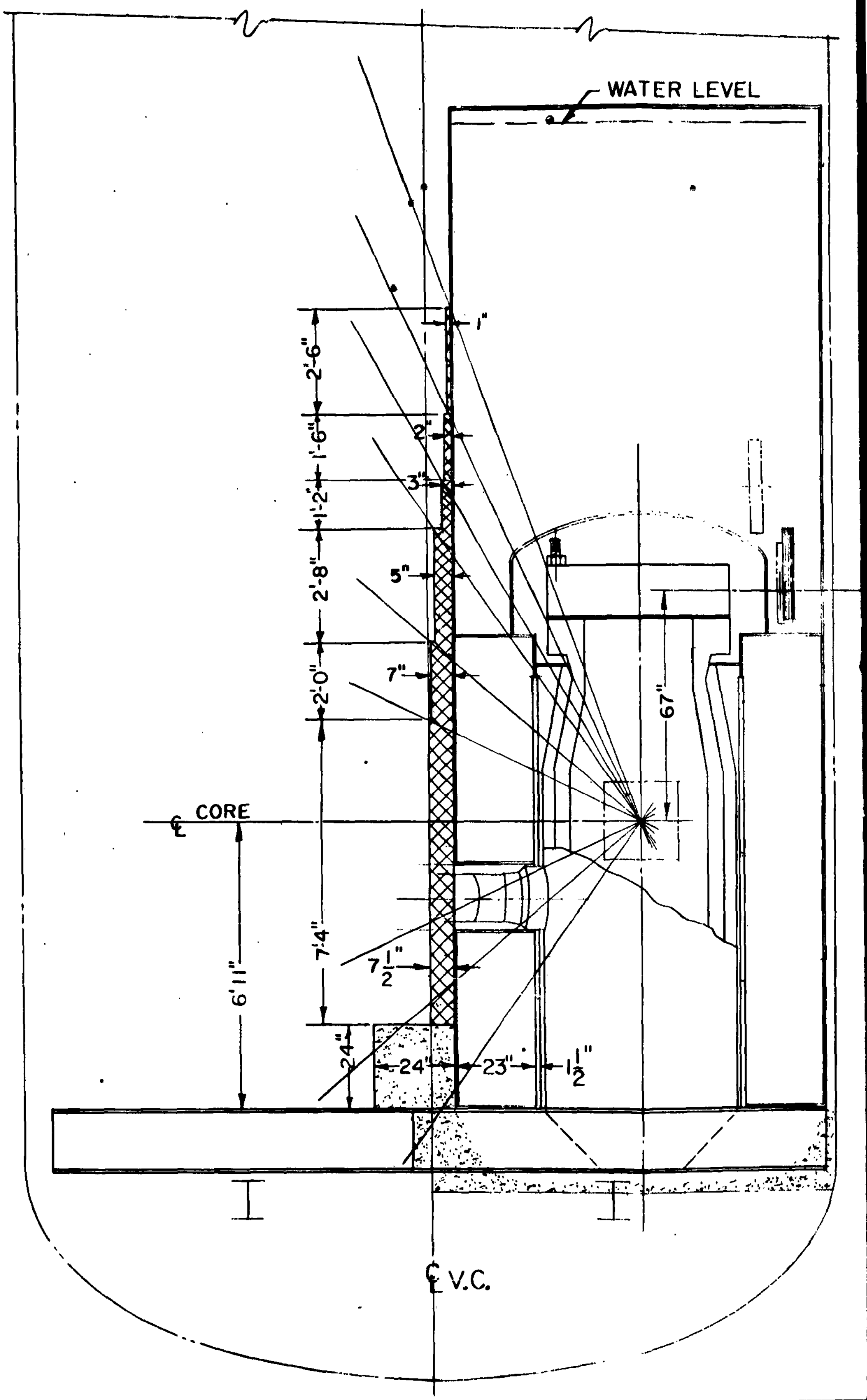


SECTION "A-A"

ALCO PRODUCTS INC

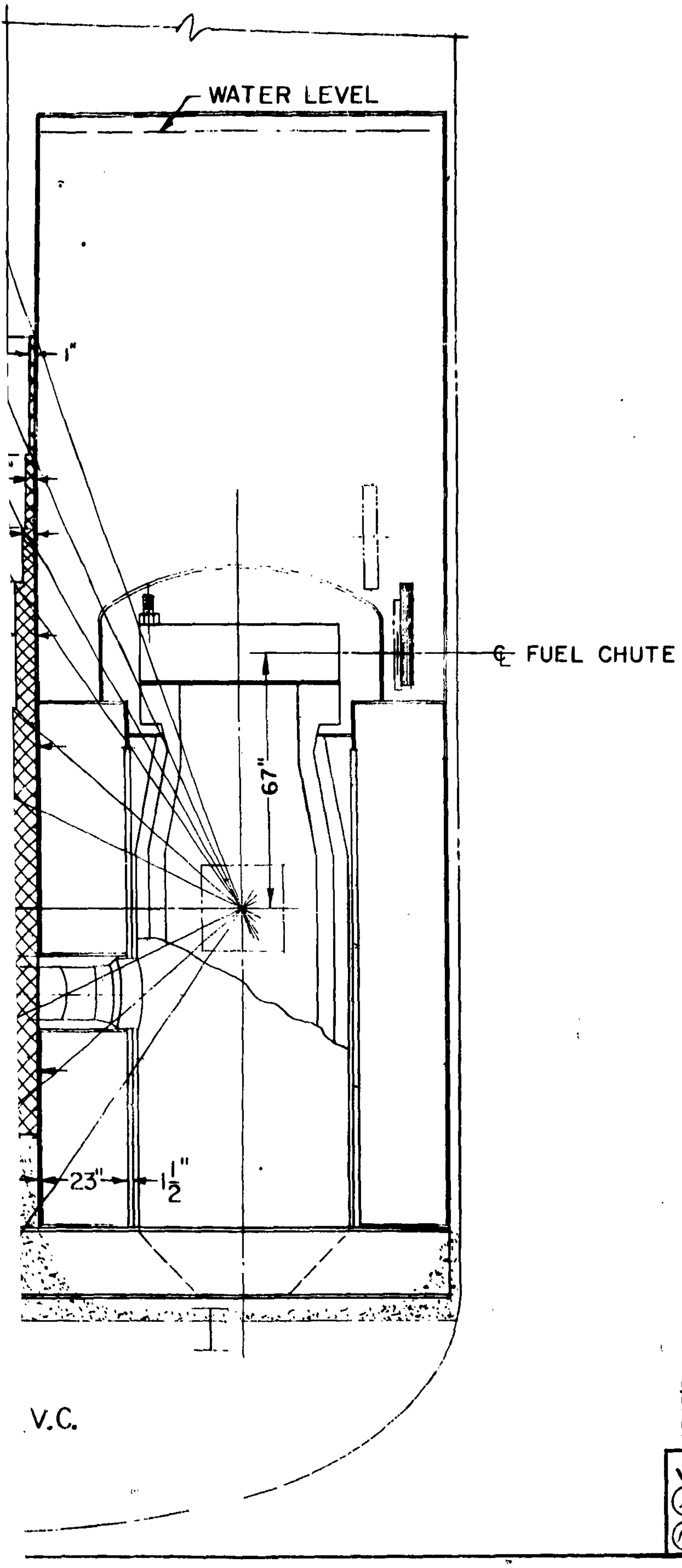
PRIMARY SHIELDING SM-2

AEL 601



WATER LEVEL
1"
2'-6"
2"
1'-6"
3"
1'-2"
5"
2'-8"
7"
2'-0"
67"
℄ CORE
7'-4"
6'-11"
7 1/2"
24"
24"
23"
1 1/2"
℄ V.C.

1



2

ALCO	**ALCO PRODUCTS, INC.** NUCLEAR POWER ENGINEERING DEPT. SCHENECTADY, N. Y., U. S. A.

SCALE 3/8"=1'0"	REF. AEL 590	DR.		6-14-60
MATERIAL SPEC.		TR.		
		CHK.		9/12/60
		APPR.		
		APPR.	JTS	9-12-60
		MET.	—	—

NAME PRIMARY SHIELDING SM 2

PART NO. AEL 602

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES.
TOLERANCES ON FINISHED FRACTIONAL DIMENSIONS TO BE ±

√ FINISH AS INDICATED IN MICROINCHES.
(/1) MACHINE FINISH - ROUGH
(/10) FLAME CUT OR SAW

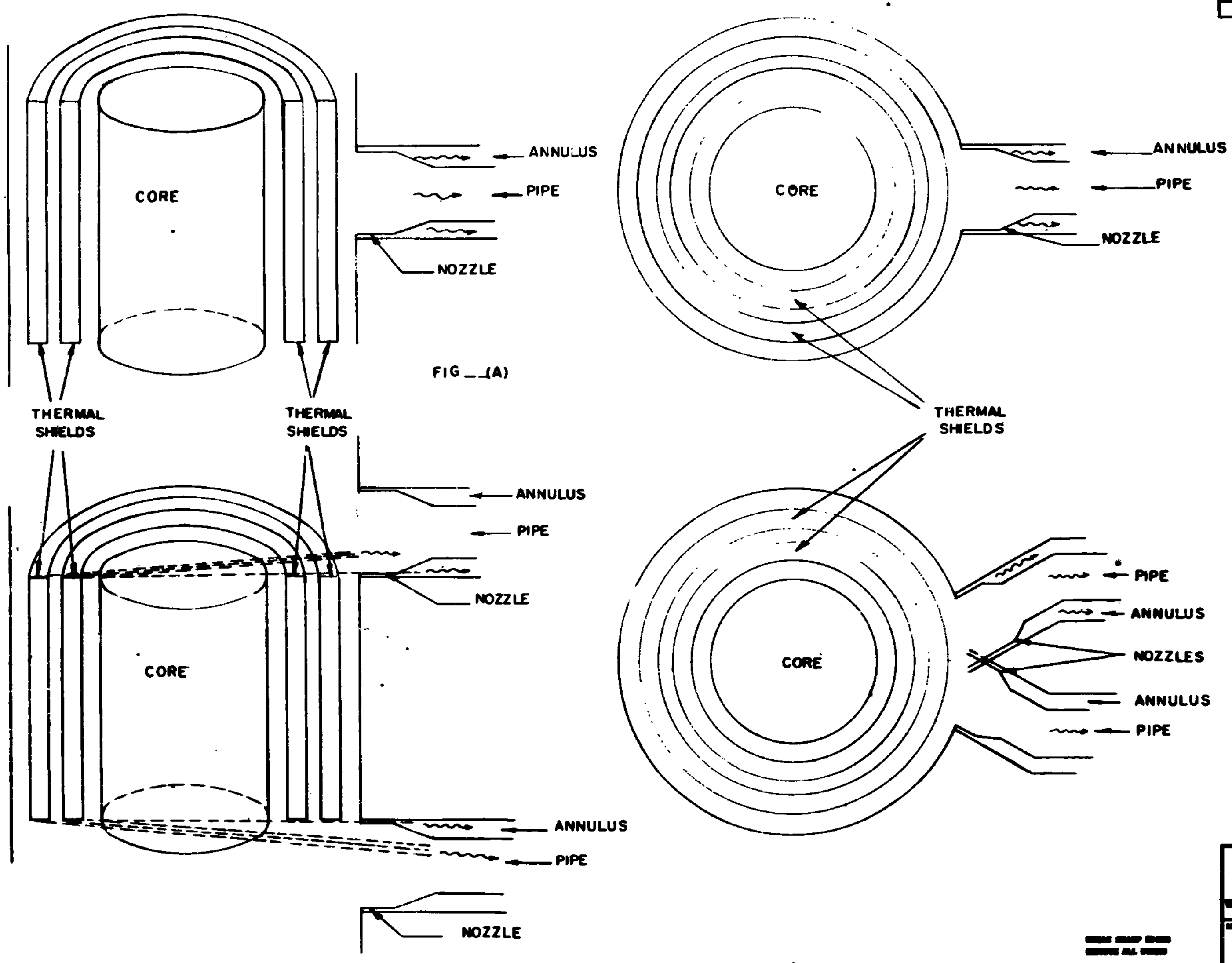


FIG __(A)

FIG __(B)

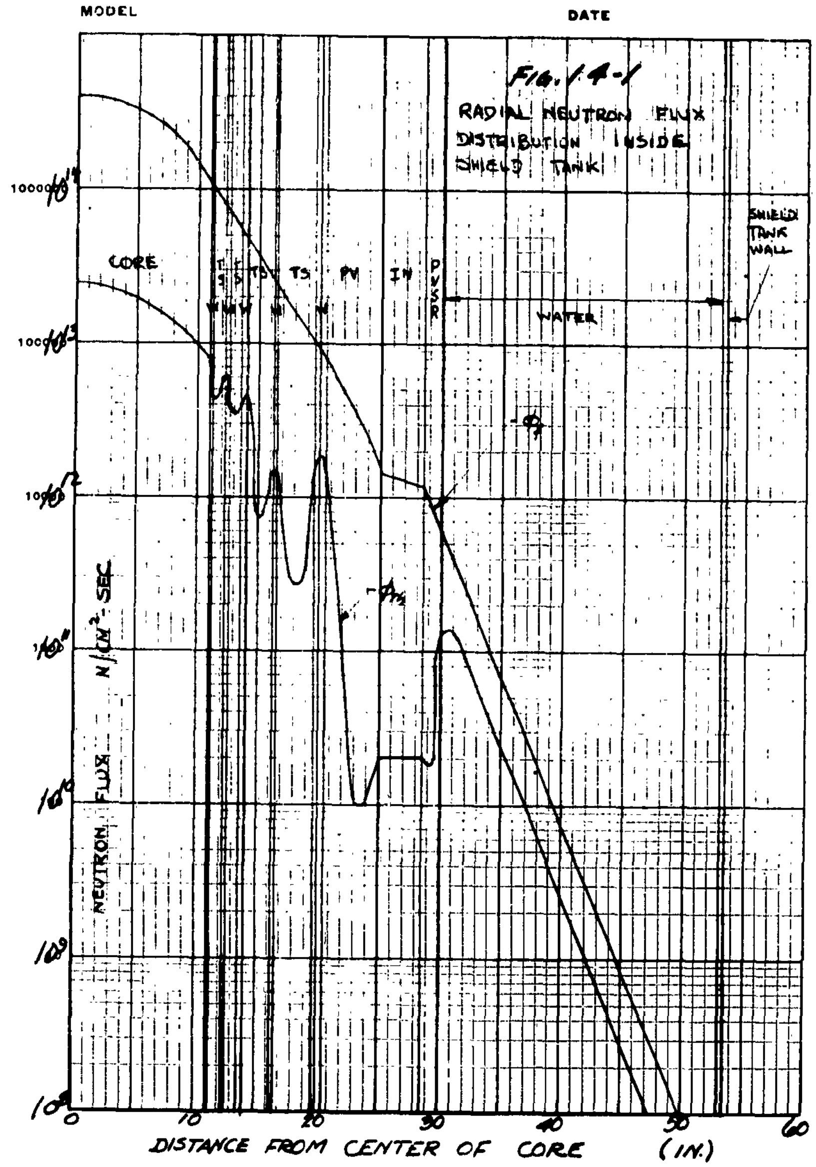
PART NO. C9-46123

ALCO PRODUCTS, INC.
SCHENECTADY, N. Y., U. S. A.

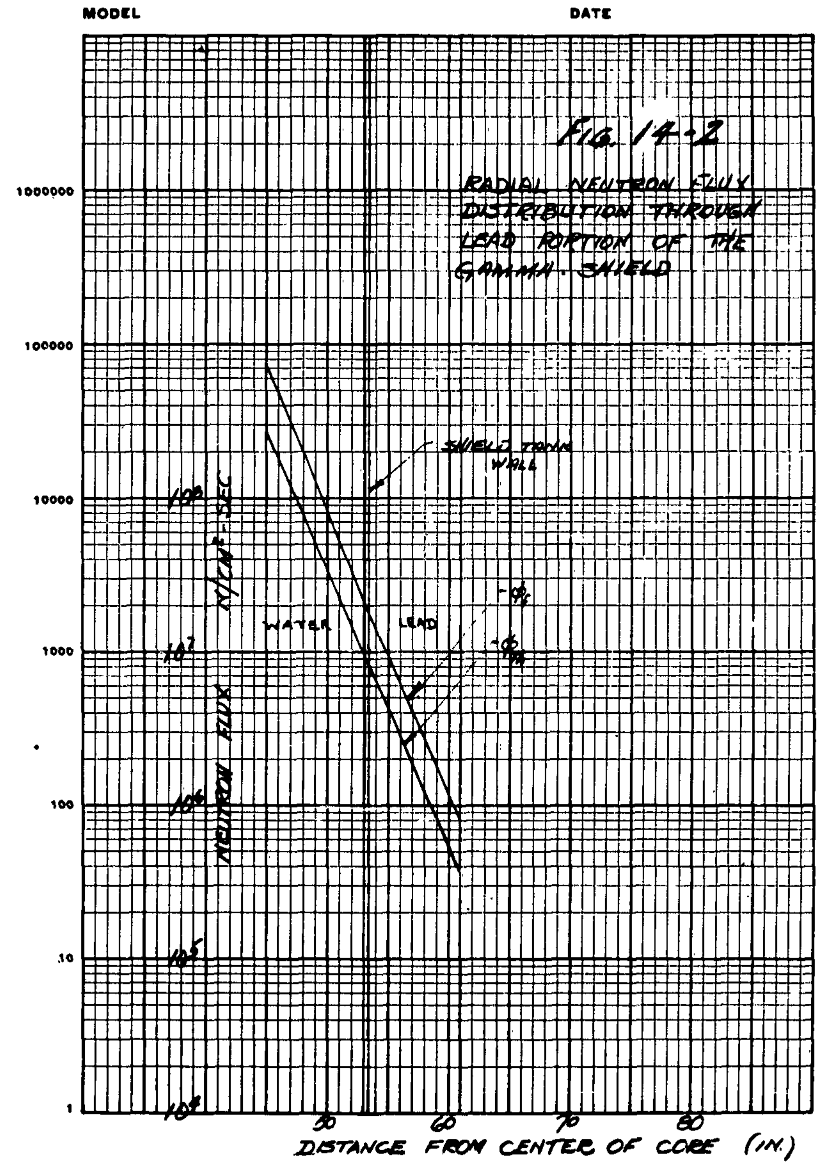
DR.	C.K.B. 8-1-60

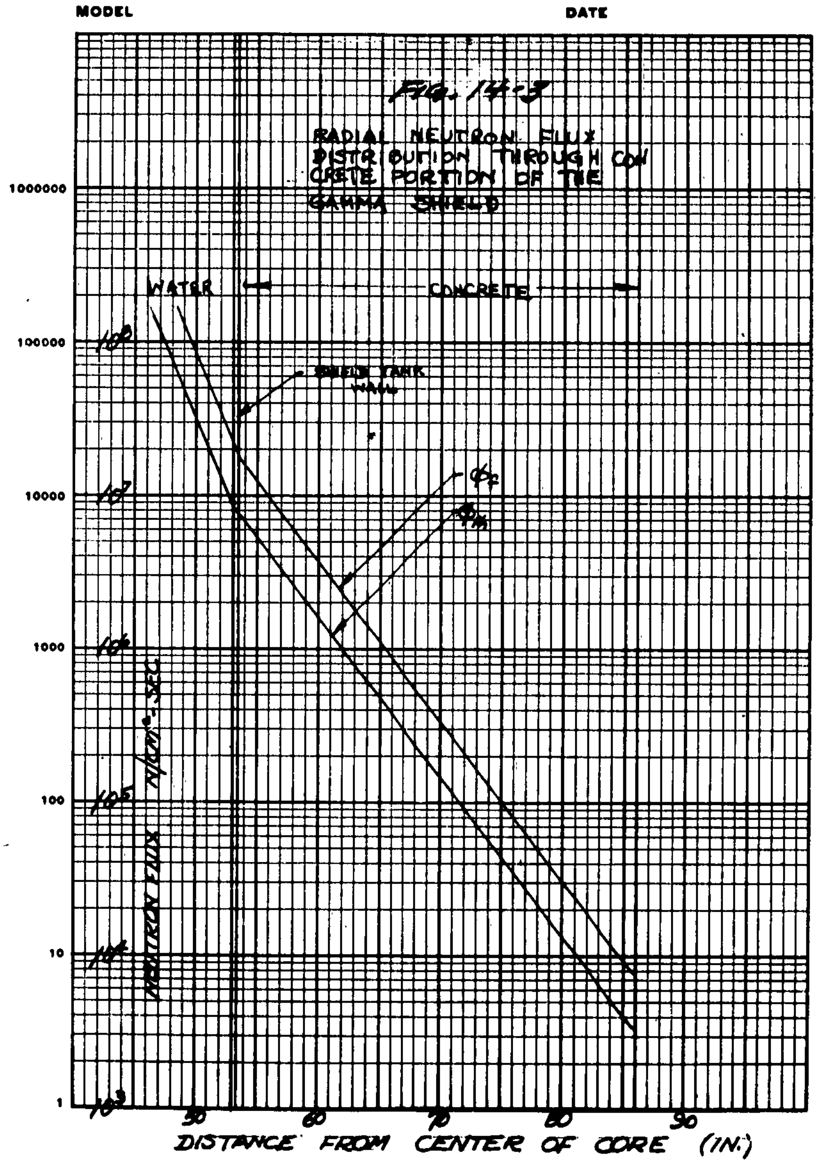
CORE & VESSEL GEOMETRY FOR SHIELD CALCULATIONS

PRINT NO. C9-46123

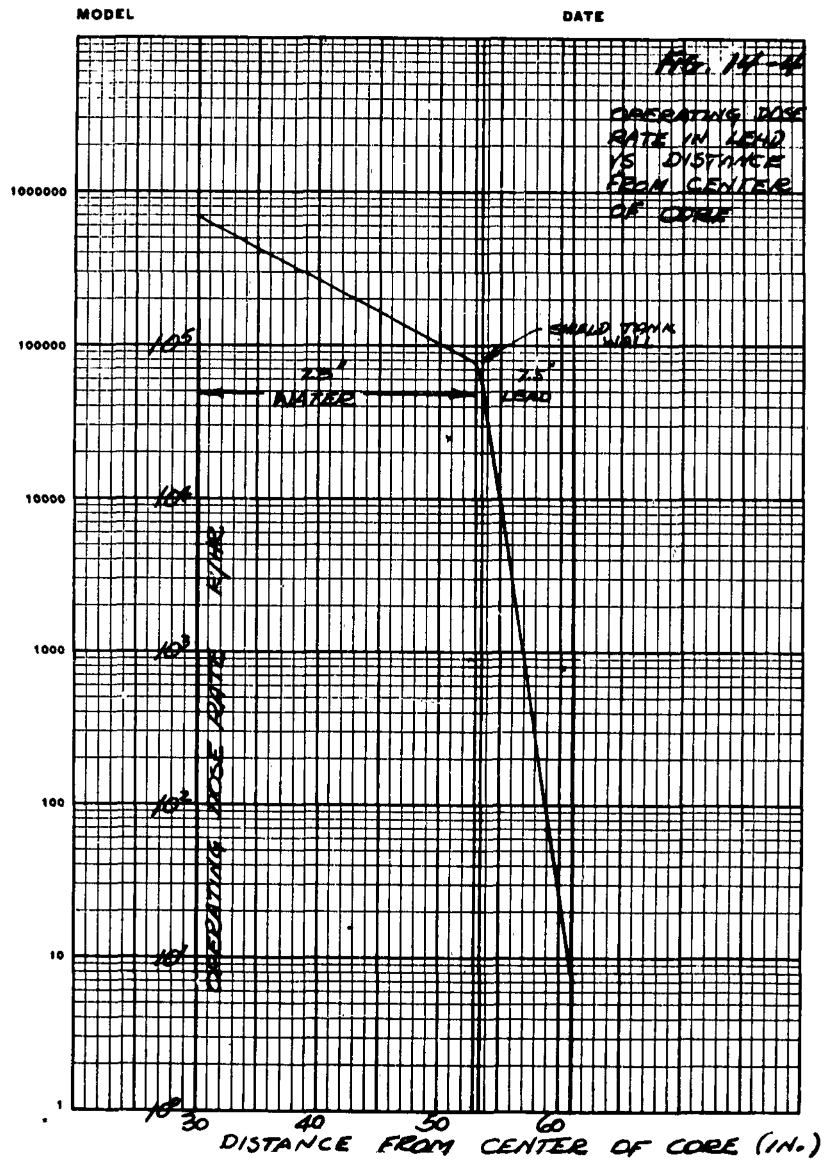


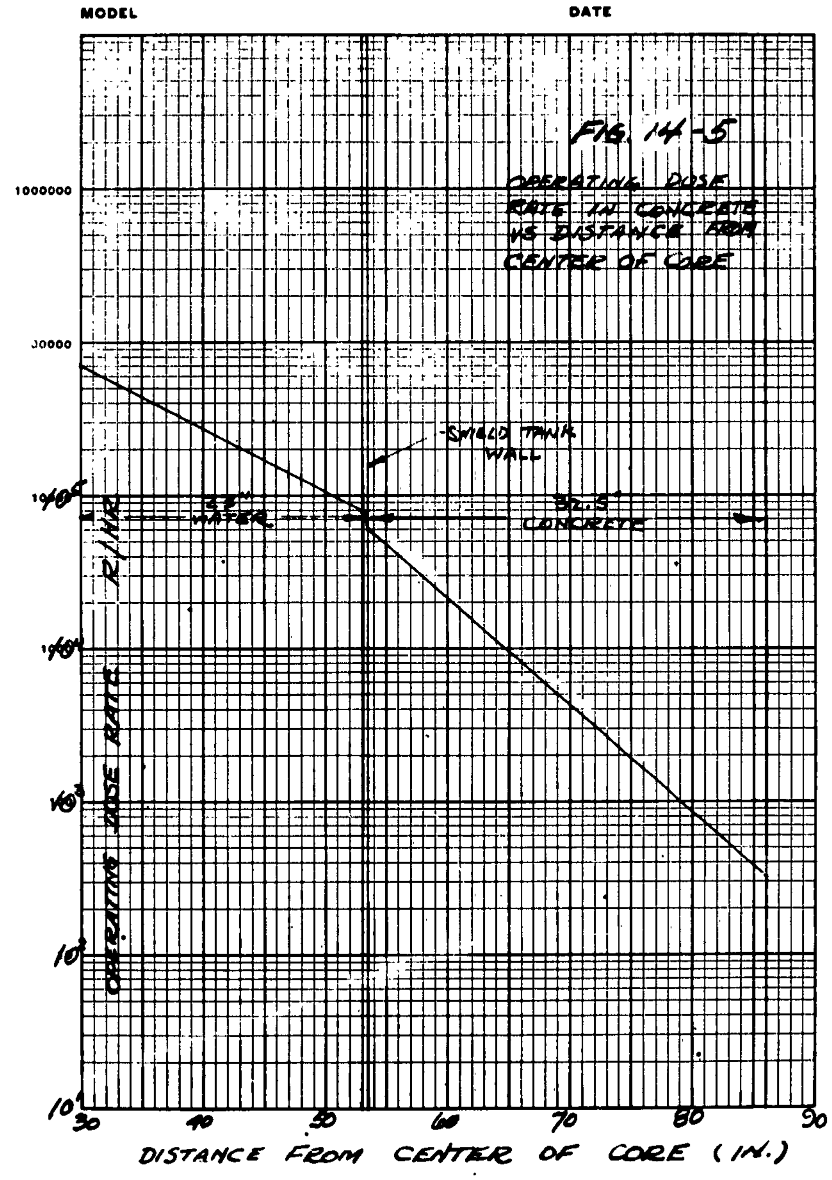
MODEL
DATE
FIG. 14-1
RADIAL NEUTRON FLUX DISTRIBUTION INSIDE SHIELD TANK
SHIELD TANK WALL
CORE
TS
TS
TS
PV
IN
P V S R
W W W
W
W
WATER
ϕ_f
ϕ_{th}
NEUTRON FLUX N/CM²-SEC
10^{14}
10^{13}
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
0
10
20
30
40
50
60
DISTANCE FROM CENTER OF CORE (IN.)

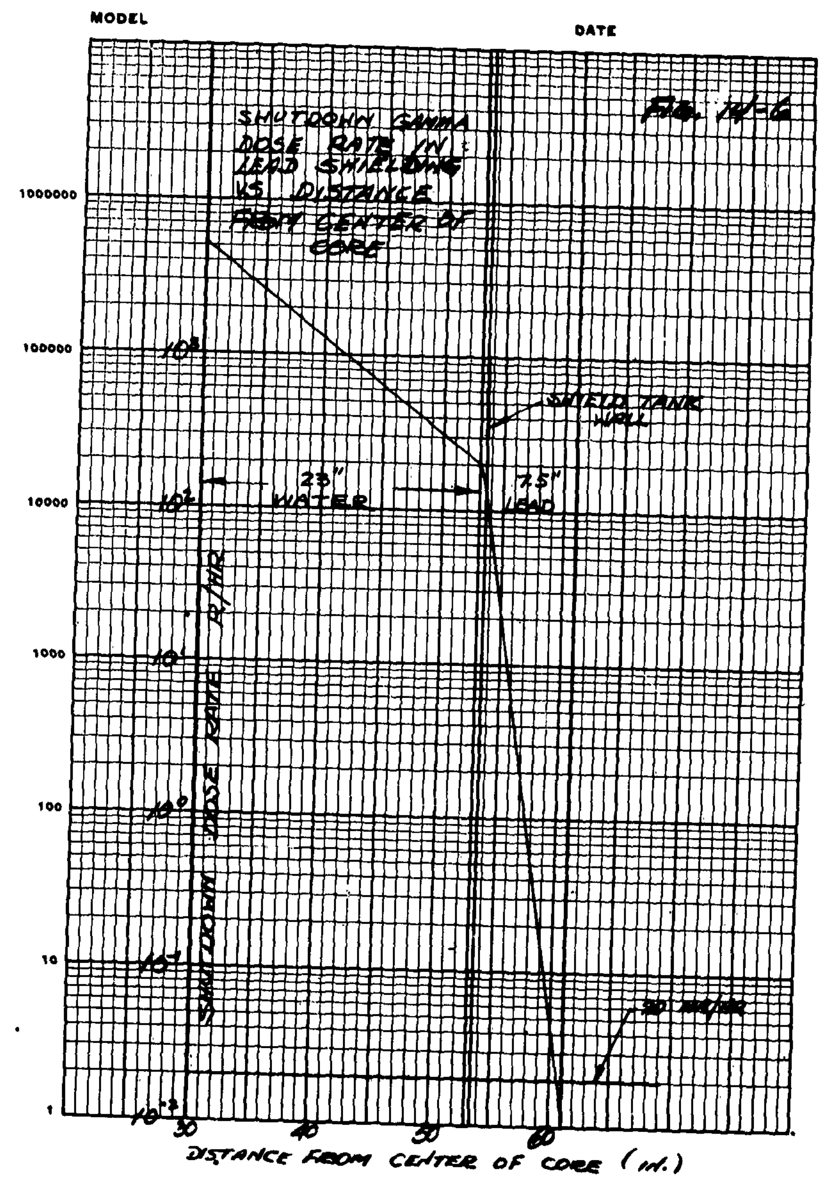

MODEL
DATE
FIG. 14-2
RADIAL NEUTRON FLUX DISTRIBUTION THROUGH LEAD PORTION OF THE GAMMA-SHIELD
SHIELD TANK WALL
WATER
LEAD
1000000
100000
10000
1000
100
10
1
10^8
10^7
10^6
10^5
10^4
NEUTRON FLUX N/CM²-SEC
50
60
70
80
DISTANCE FROM CENTER OF CORE (IN.)



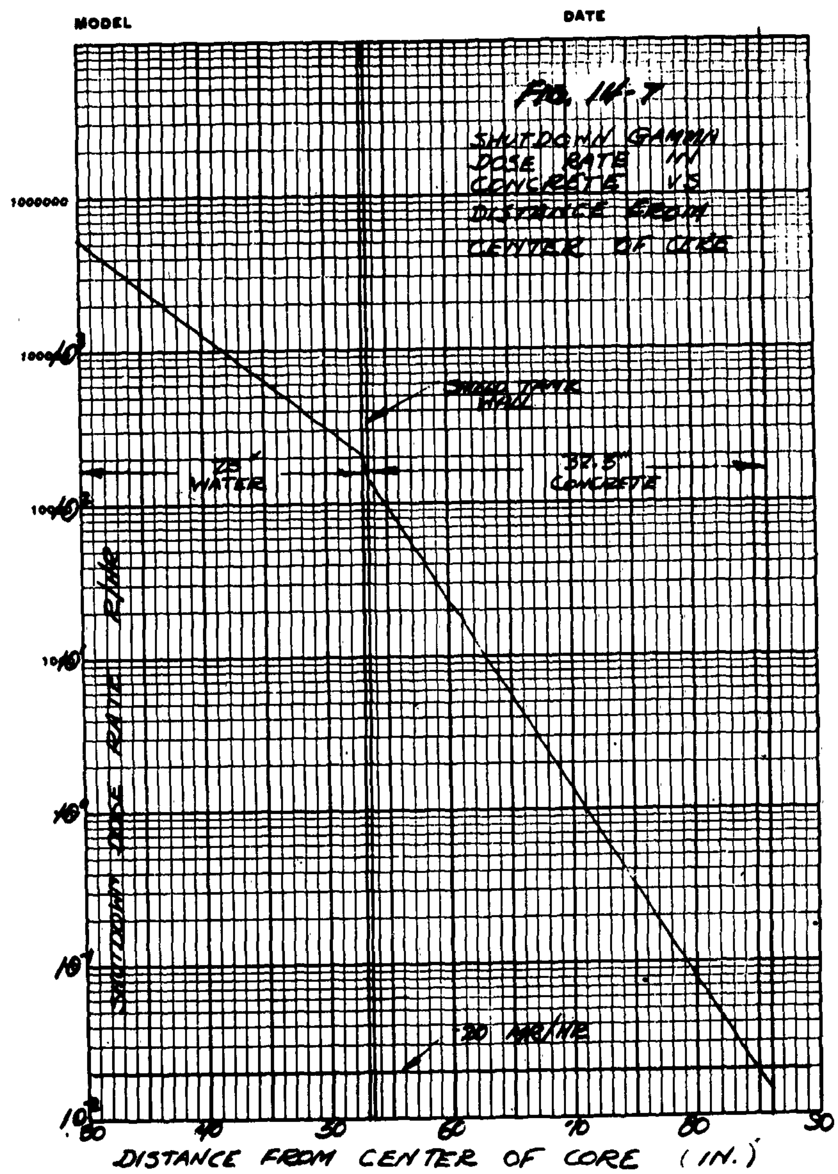
MODEL
DATE
FIG. 14-3
RADIAL NEUTRON FLUX DISTRIBUTION THROUGH CONCRETE PORTION OF THE GAMMA SHIELD
WATER
CONCRETE
SHIELD TANK WALL
ϕ_F
ϕ_{th}
NEUTRON FLUX N/CM²-SEC
10^8
10^7
10^6
10^5
10^4
10^3
1000000
100000
10000
1000
100
10
1
50
60
70
80
90
DISTANCE FROM CENTER OF CORE (IN.)


MODEL
DATE
FIG. 14-4
OPERATING DOSE RATE IN LEAD VS DISTANCE FROM CENTER OF CORE
SHIELD TANK WALL
23"
WATER
7.5"
LEAD
OPERATING DOSE RATE R/HR
1000000
100000
10000
1000
100
10
1
10^5
10^4
10^3
10^2
10^1
10^0
30
40
50
60
DISTANCE FROM CENTER OF CORE (IN.)


MODEL
DATE
FIG. 14-5
OPERATING DOSE RATE IN CONCRETE VS DISTANCE FROM CENTER OF CORE
SHIELD TANK WALL
23" WATER
32.5" CONCRETE
OPERATING DOSE RATE R/HR
DISTANCE FROM CENTER OF CORE (IN.)



MODEL
DATE
FIG. IV-6
SHUTDOWN GAMMA DOSE RATE IN LEAD SHIELDING VS DISTANCE FROM CENTER OF CORE
1000000
100000
10000
1000
100
10
1
10^3
10^2
10^1
10^0
10^{-1}
10^{-2}
SHIELD TANK WALL
23" WATER
7.5" LEAD
SHUTDOWN DOSE RATE R/HR
30
40
50
60
DISTANCE FROM CENTER OF CORE (in.)


MODEL
DATE
FIG. 14-7
SHUTDOWN GAMMA DOSE RATE IN CONCRETE VS DISTANCE FROM CENTER OF CORE
1000000
SHIELD TANK WALL
28" WATER
32.5" CONCRETE
SHUTDOWN DOSE RATE R/HR
20 MR/HR
30
40
50
60
70
80
90
DISTANCE FROM CENTER OF CORE (IN.)

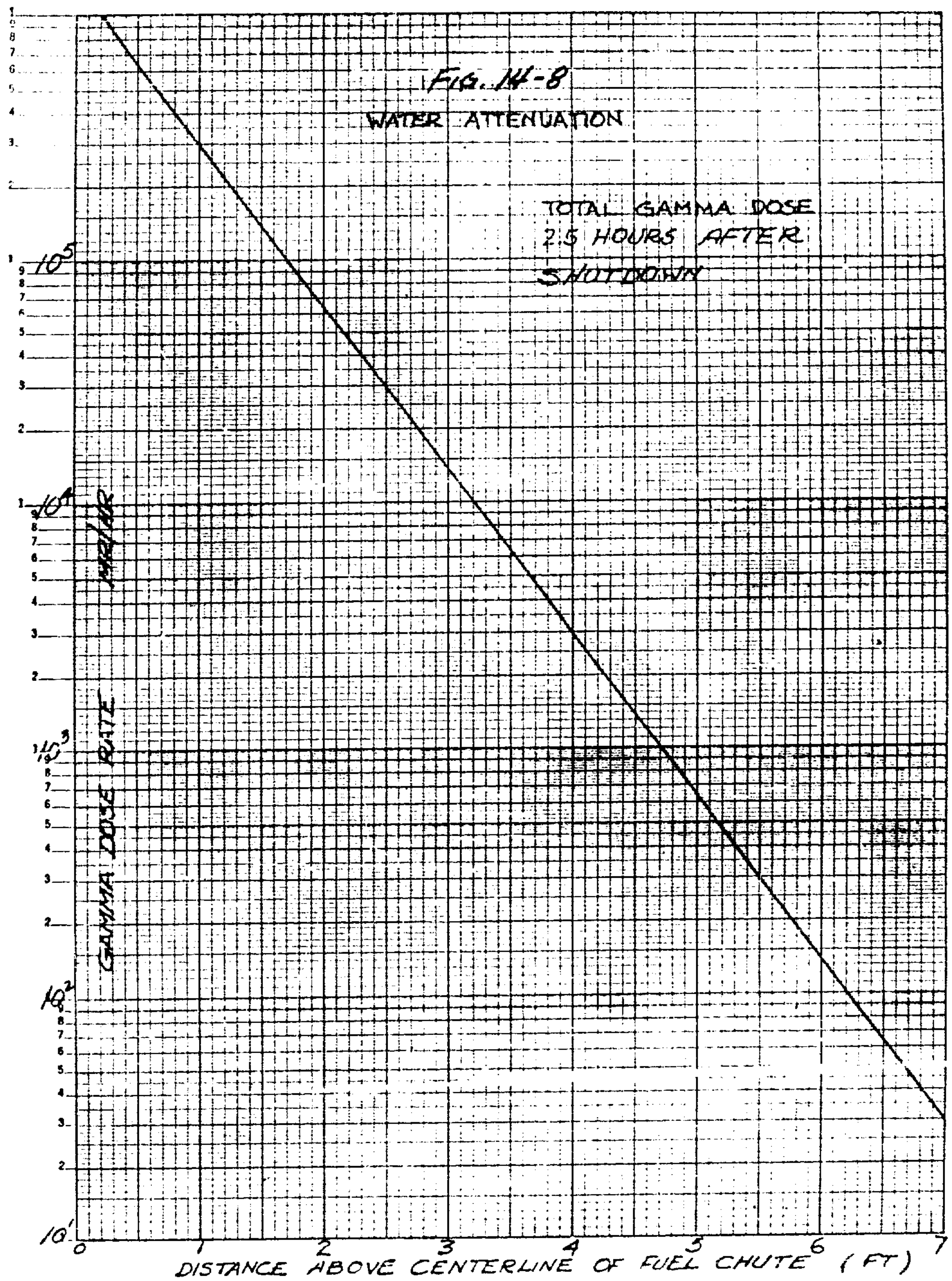


FIG. 14-8
WATER ATTENUATION
TOTAL GAMMA DOSE
2.5 HOURS AFTER
SHUTDOWN
GAMMA DOSE RATE MR/HR
10^5
10^4
10^3
10^2
10^1
0
1
2
3
4
5
6
7
DISTANCE ABOVE CENTERLINE OF FUEL CHUTE (FT)

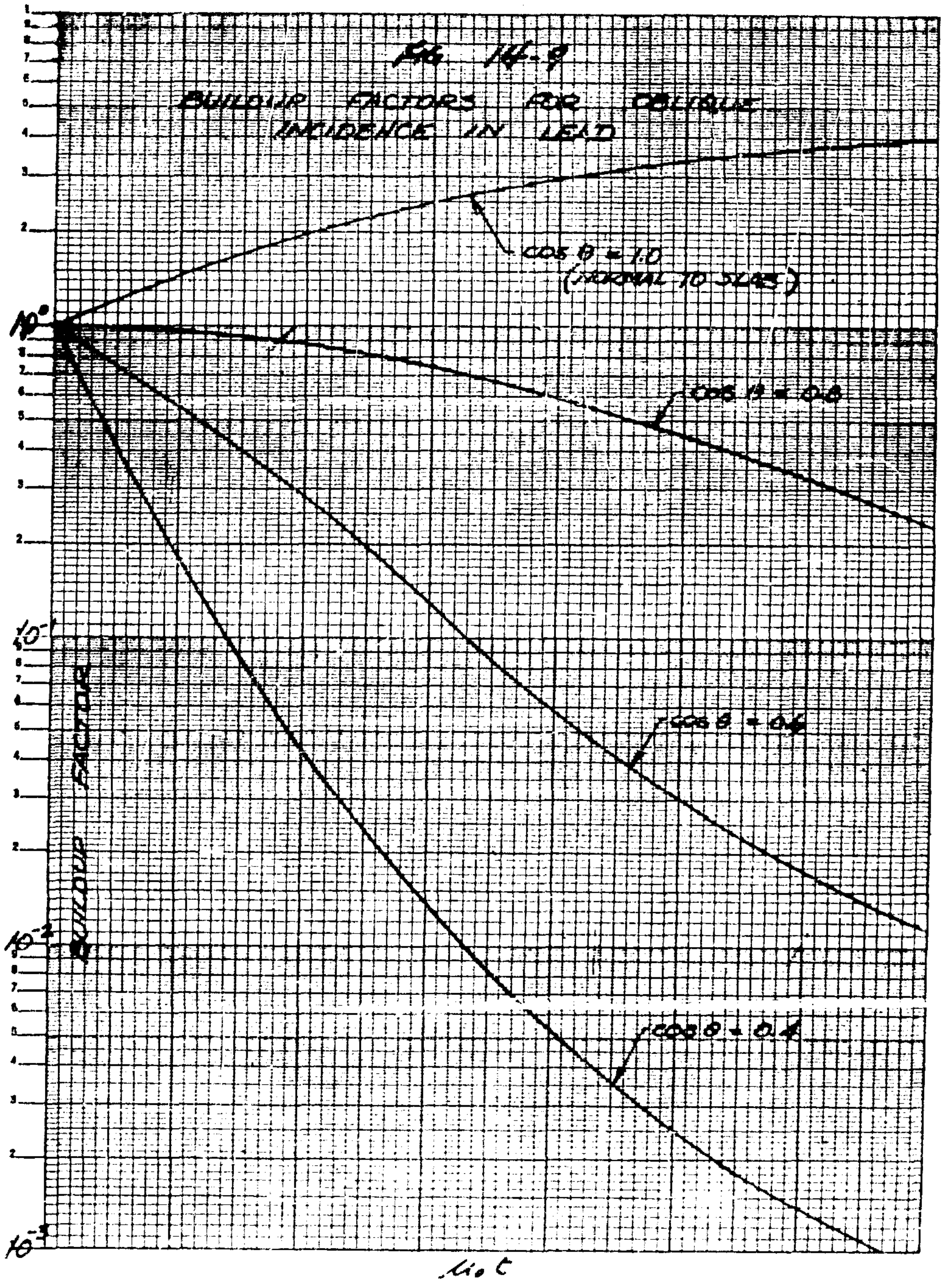
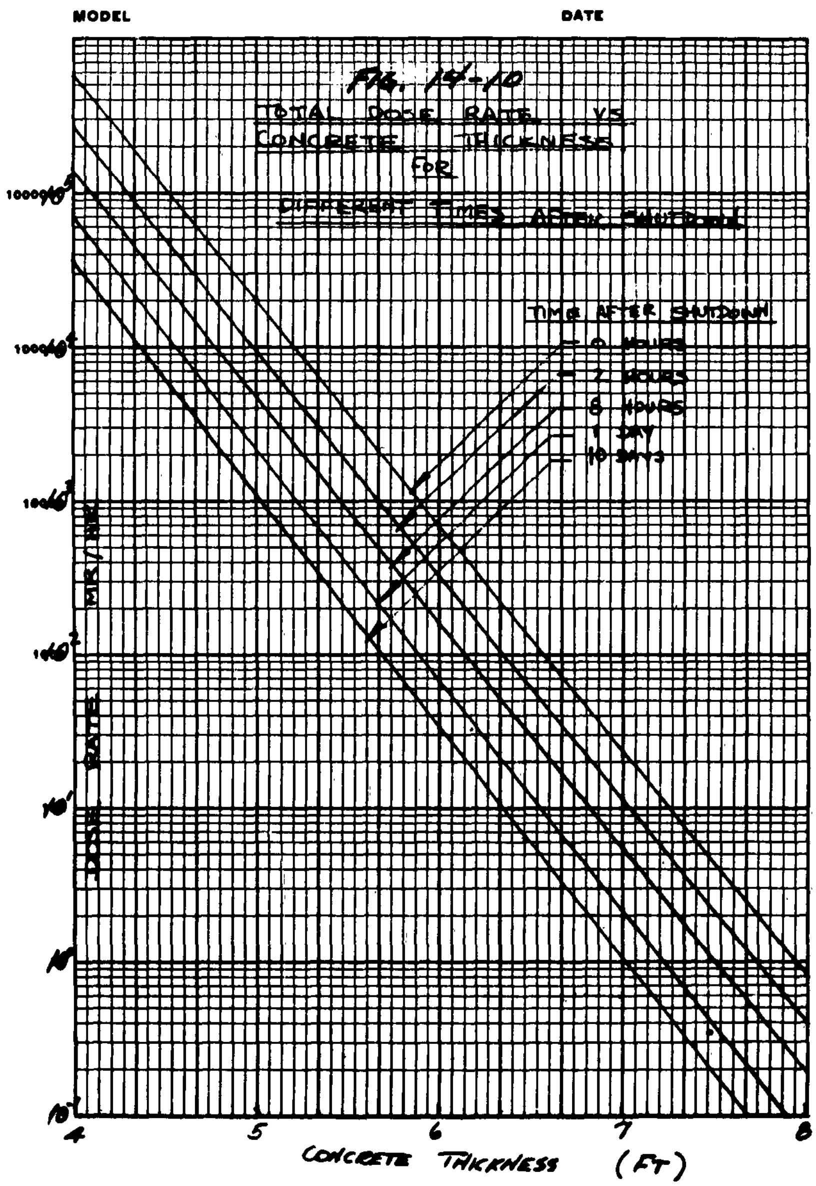


FIG. 14-9
BUILDUP FACTORS FOR OBLIQUE INCIDENCE IN LEAD
cos θ = 1.0
(NORMAL TO SLAB)
cos θ = 0.8
cos θ = 0.6
cos θ = 0.4
BUILDUP FACTOR
10^0
10^{-1}
10^{-2}
10^{-3}
$\mu_0 t$


MODEL
DATE
FIG. 14-10
TOTAL DOSE RATE VS
CONCRETE THICKNESS
FOR
DIFFERENT TIMES AFTER SHUTDOWN
TIME AFTER SHUTDOWN
0 HOURS
2 HOURS
8 HOURS
1 DAY
10 DAYS
DOSE RATE MR/HR
10^5
10^4
10^3
10^2
10^1
10^0
10^{-1}
4
5
6
7
8
CONCRETE THICKNESS (FT)

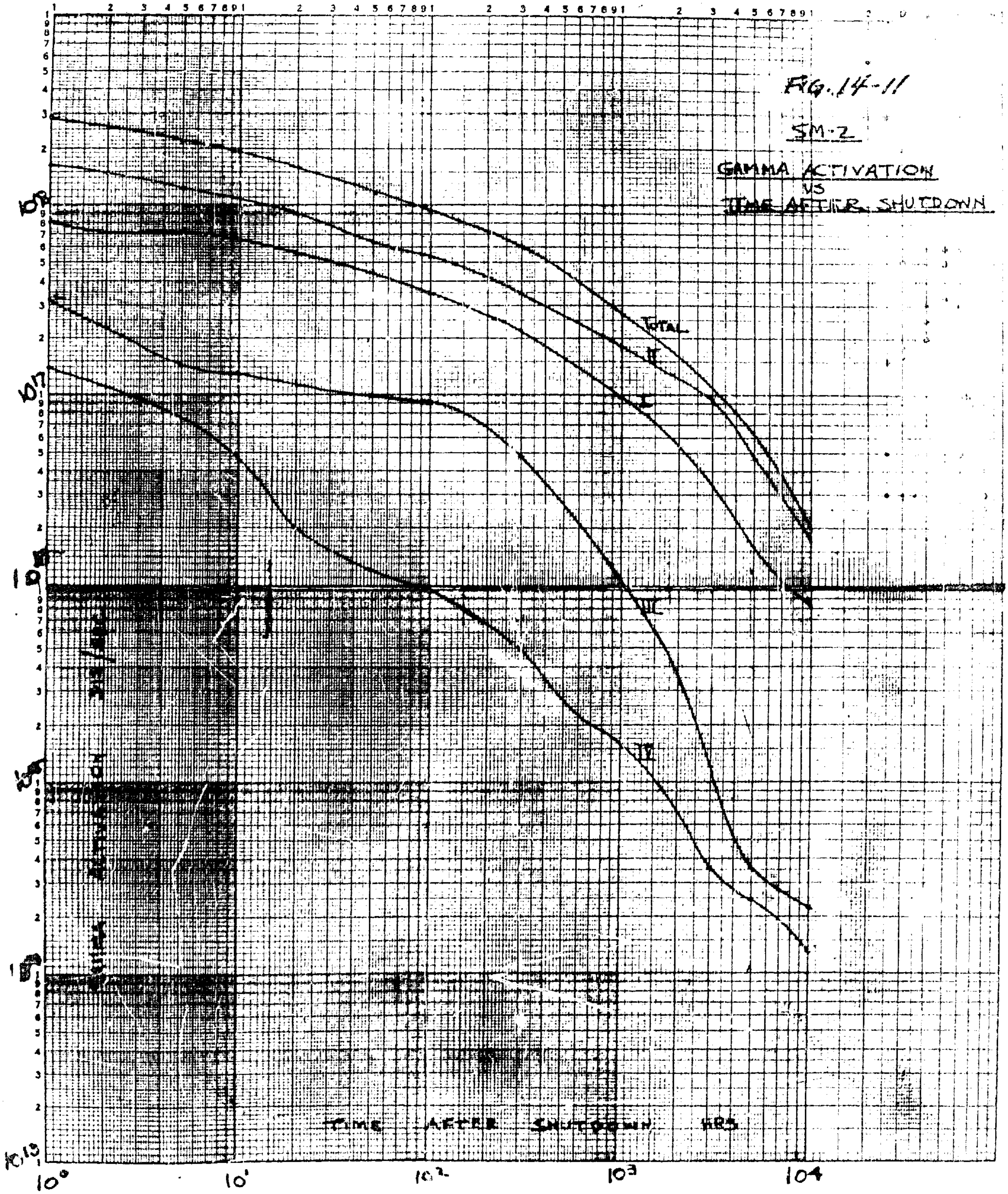


FIG. 14-11
SM-2
GAMMA ACTIVATION
VS
TIME AFTER SHUTDOWN
TOTAL
II
I
III
IV
TIME AFTER SHUTDOWN HRS
10^{18}
10^{17}
10^{13}
10^0
10^1
10^2
10^3
10^4

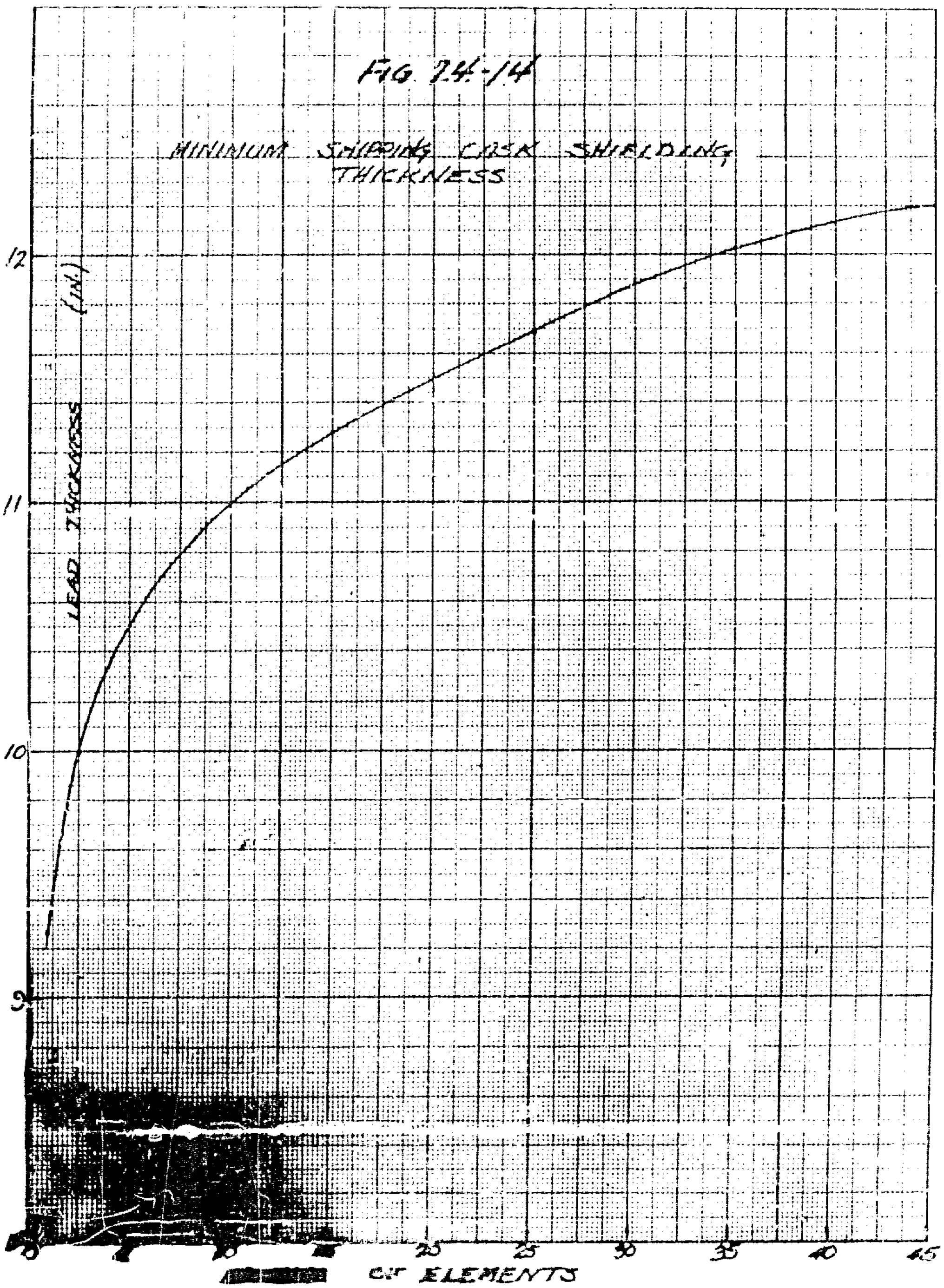


FIG. 24-14
MINIMUM SHIPPING CASK SHIELDING THICKNESS
LEAD THICKNESS (IN.)
12
11
10
9
10
20
25
30
35
40
45
[redacted] OF ELEMENTS

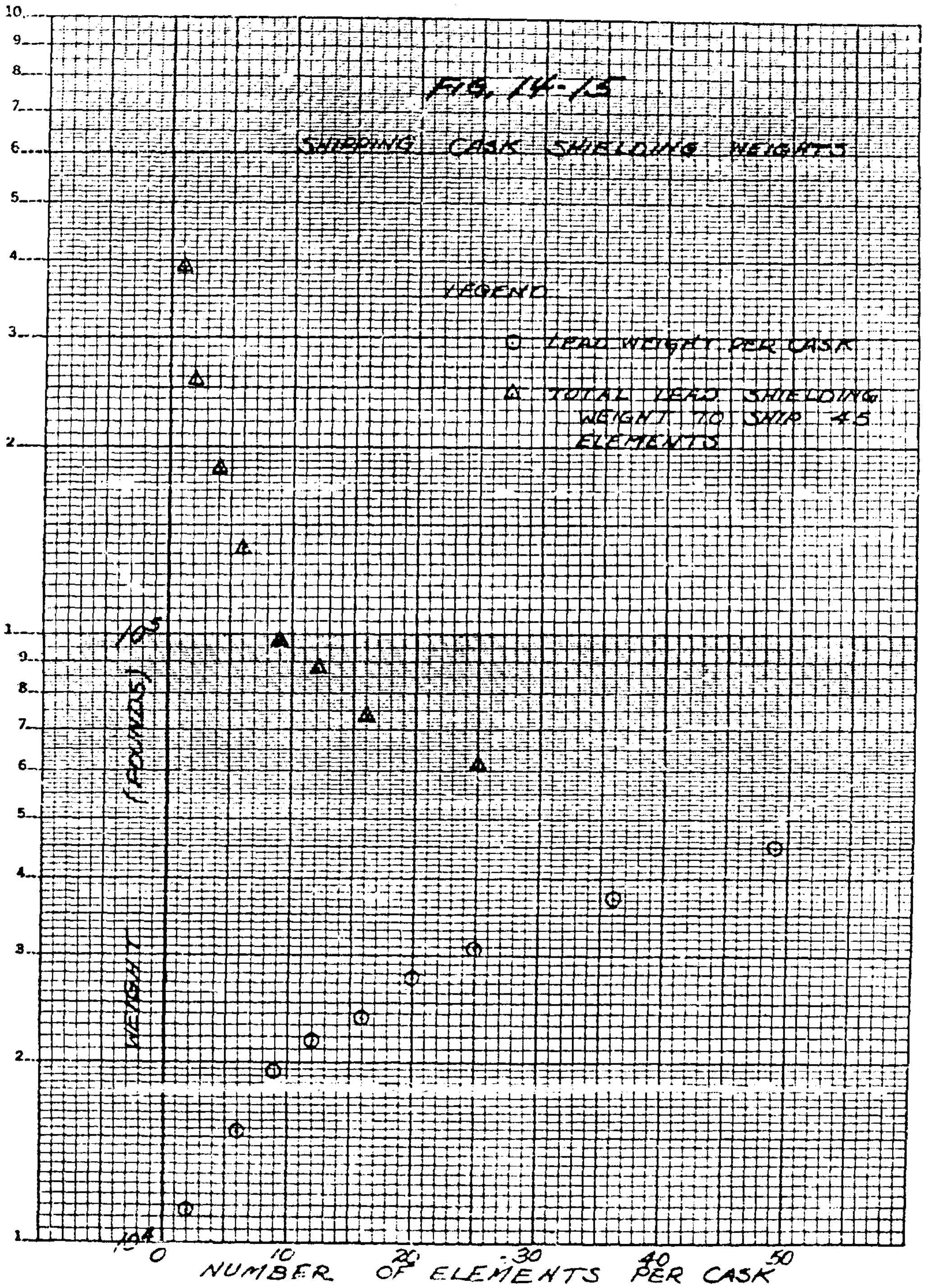


FIG. 14-15
SHIPPING CASK SHIELDING WEIGHTS
LEGEND
⊙ LEAD WEIGHT PER CASK
△ TOTAL LEAD SHIELDING WEIGHT TO SHIP 45 ELEMENTS
WEIGHT (POUNDS)
10^5
10^4
0
10
20
30
40
50
NUMBER OF ELEMENTS PER CASK

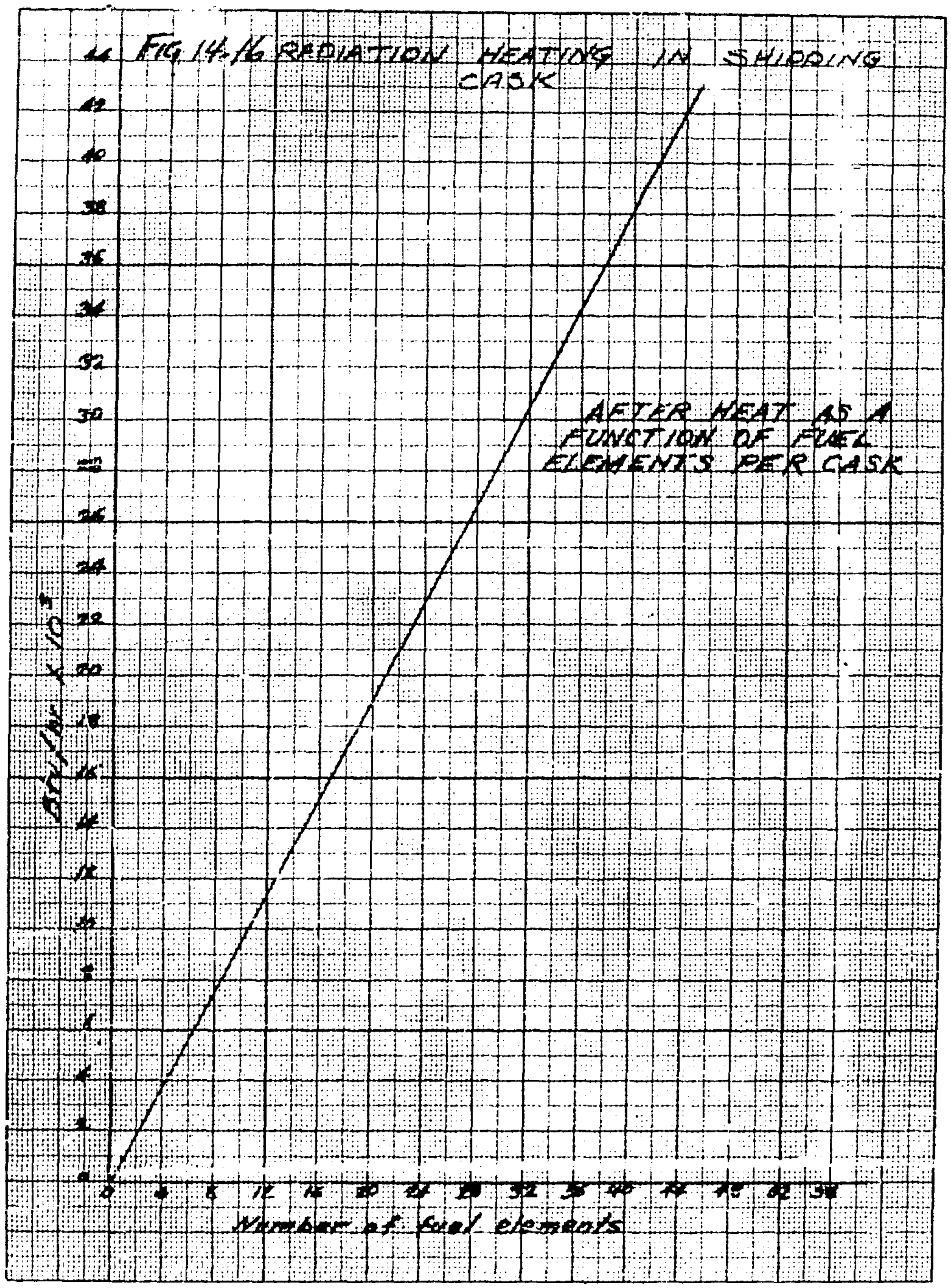


FIG. 14-16 RADIATION HEATING IN SHIPPING CASK
AFTER HEAT AS A FUNCTION OF FUEL ELEMENTS PER CASK
Btu/hr X 10^3
Number of fuel elements

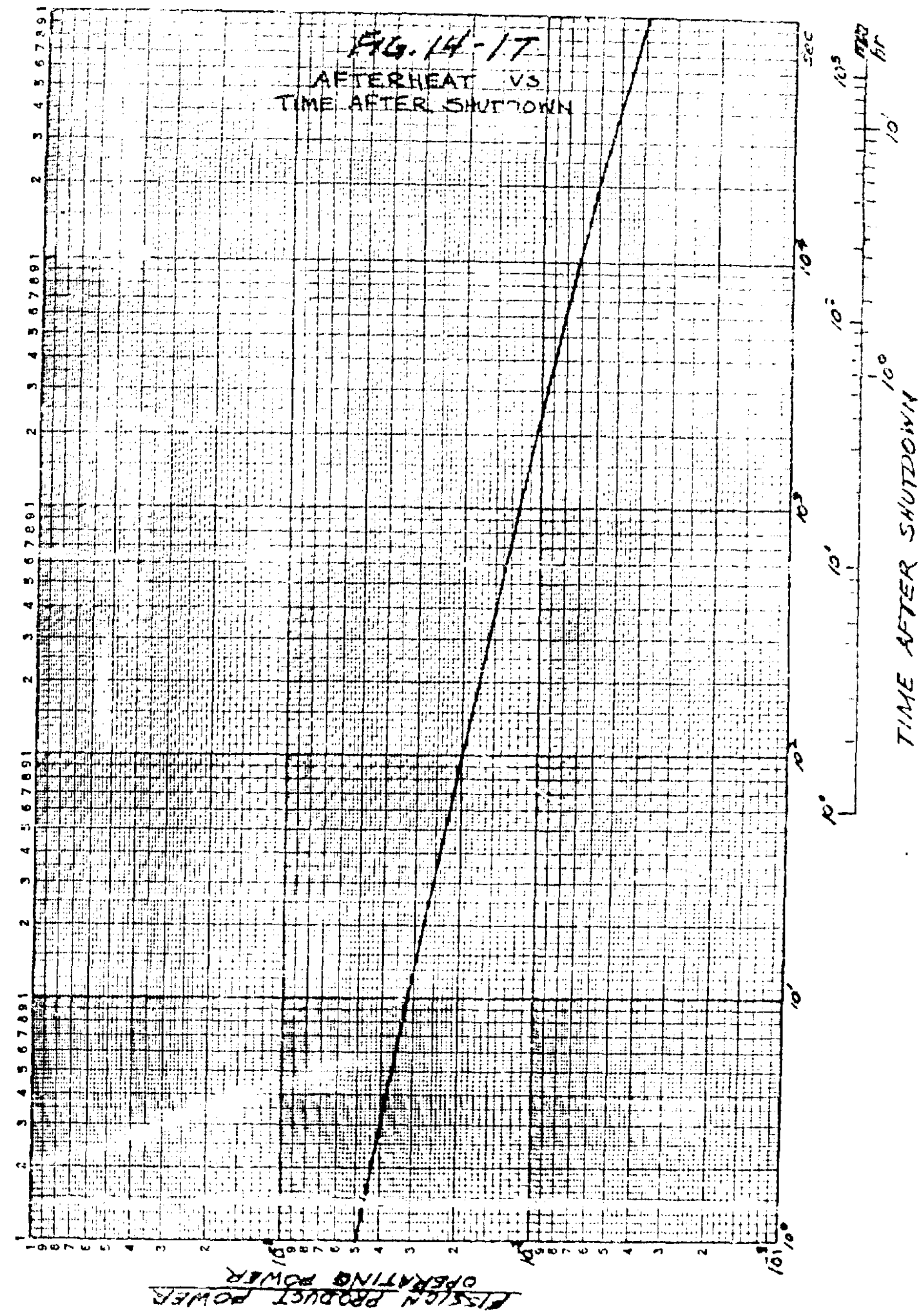


FIG. 14-17
AFTERHEAT VS
TIME AFTER SHUTDOWN
TIME AFTER SHUTDOWN
sec
hr
FISSION PRODUCT POWER
OPERATING POWER

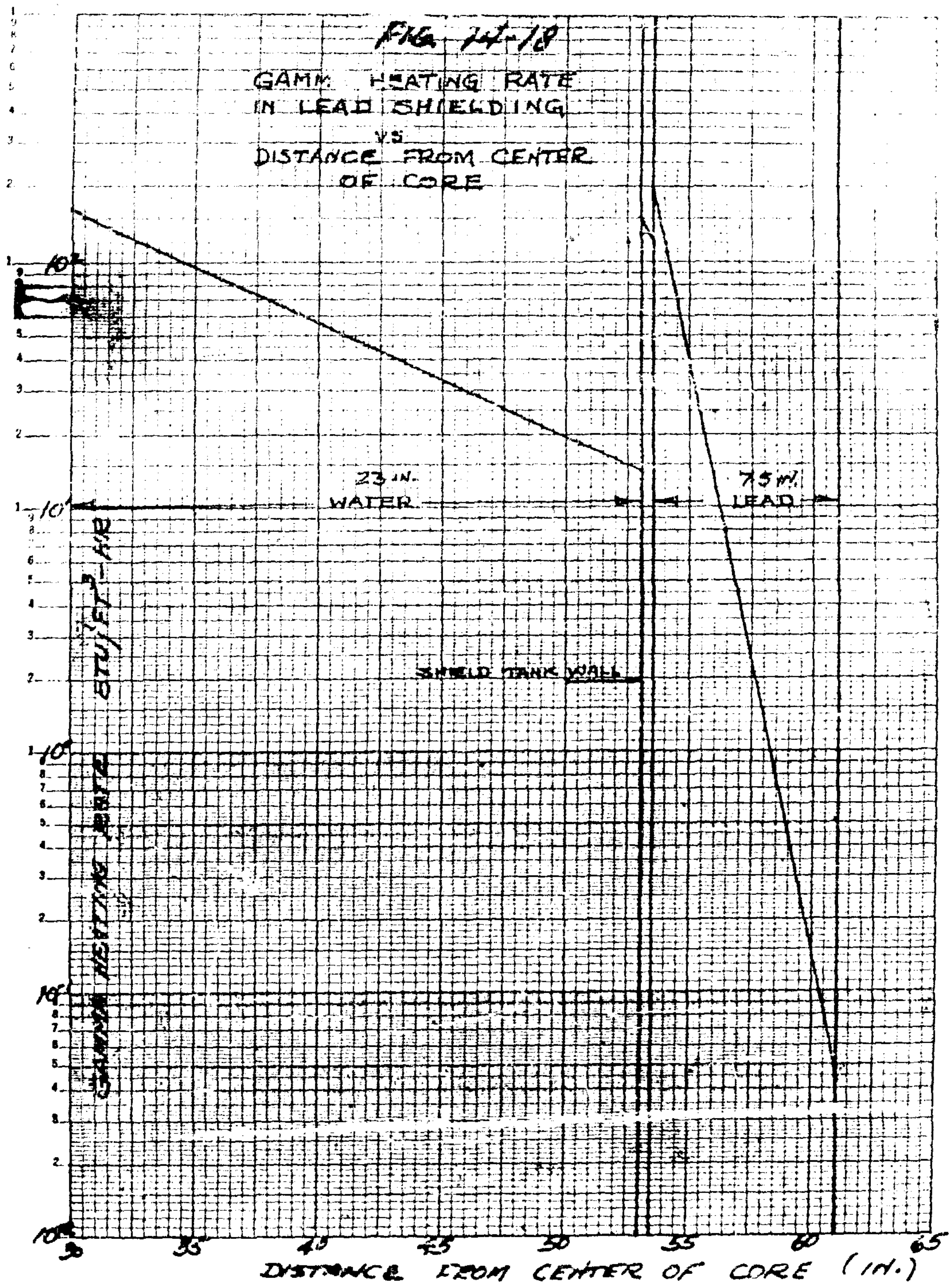


FIG. 24-18
GAMMA HEATING RATE
IN LEAD SHIELDING
VS
DISTANCE FROM CENTER
OF CORE
23 IN.
WATER
7.5 IN.
LEAD
SHIELD TANK WALL
GAMMA HEATING RATE BTU/FT³-HR
DISTANCE FROM CENTER OF CORE (IN.)

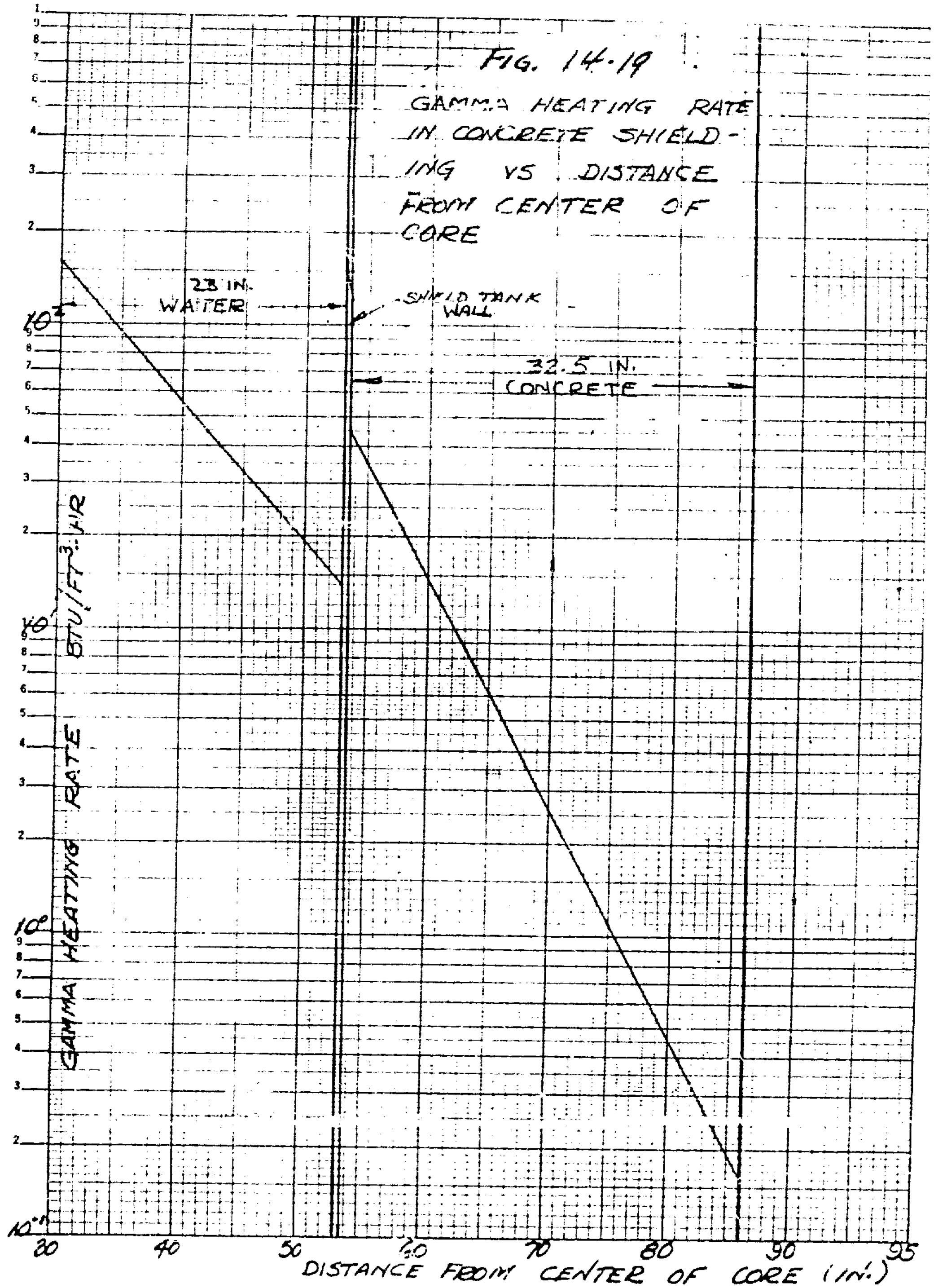


FIG. 14.19
GAMMA HEATING RATE IN CONCRETE SHIELDING VS DISTANCE FROM CENTER OF CORE
23 IN. WATER
SHIELD TANK WALL
32.5 IN. CONCRETE
GAMMA HEATING RATE BTU/FT³·HR
DISTANCE FROM CENTER OF CORE (IN.)

FIG 14-20

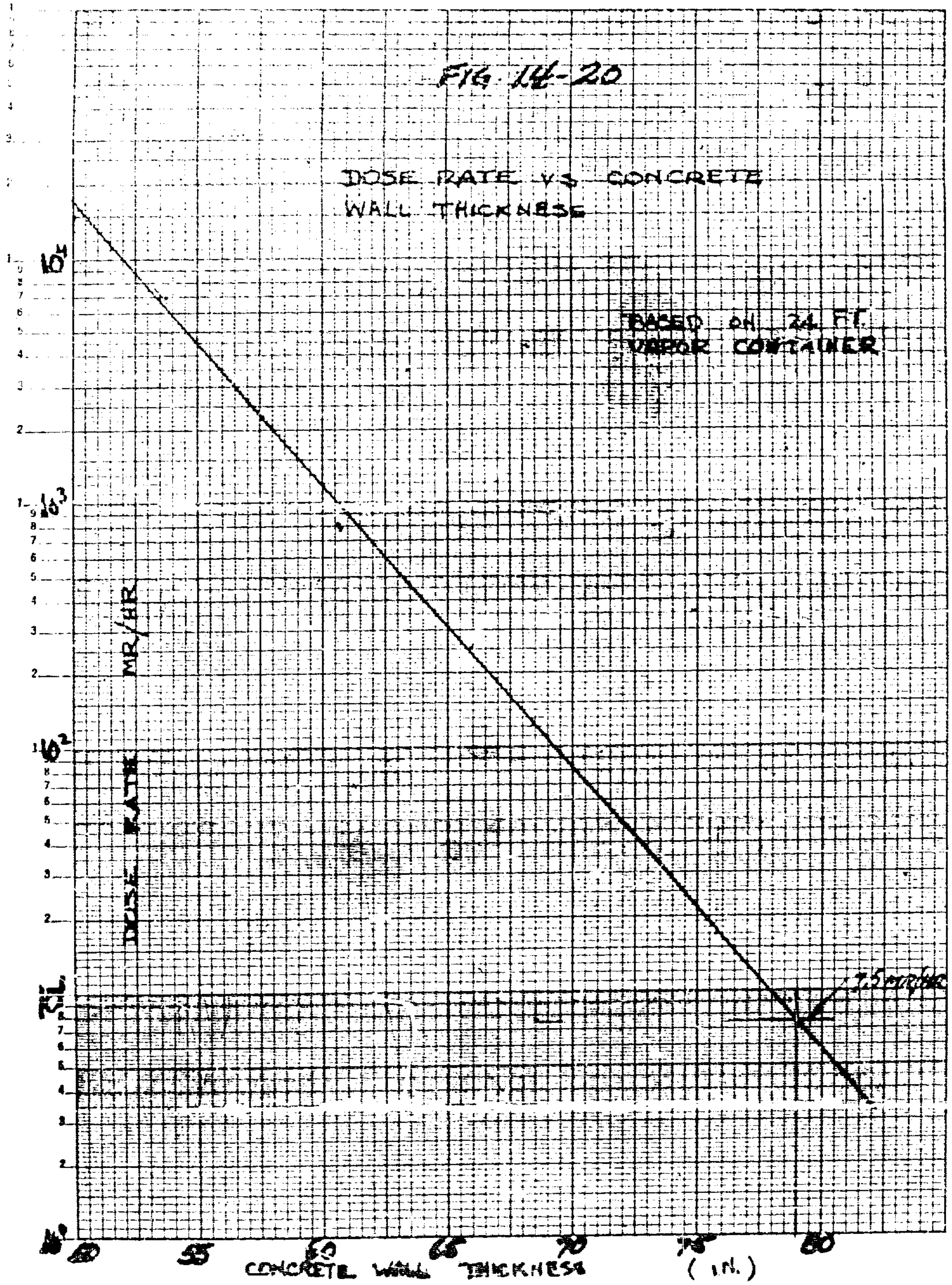
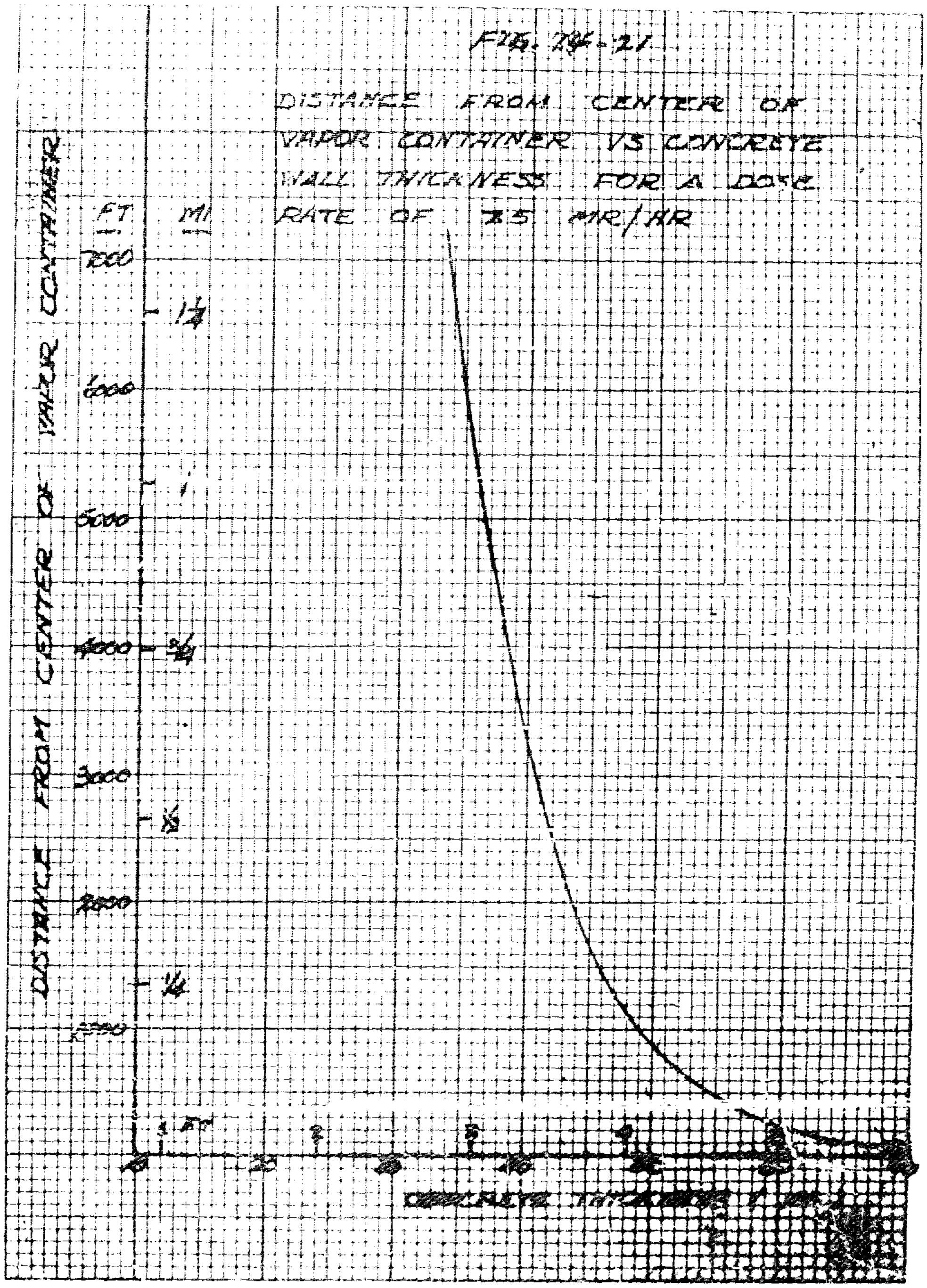

DOSE RATE VS CONCRETE WALL THICKNESS
BASED ON 24 FT. VAPOR CONTAINER
7.5 MR/HR
DOSE RATE MR/HR
CONCRETE WALL THICKNESS (IN.)
50
55
60
65
70
75
80

FIG. 24-21

DISTANCE FROM CENTER OF VAPOR CONTAINER VS CONCRETE WALL THICKNESS FOR A DOSE RATE OF 2.5 MR/HR



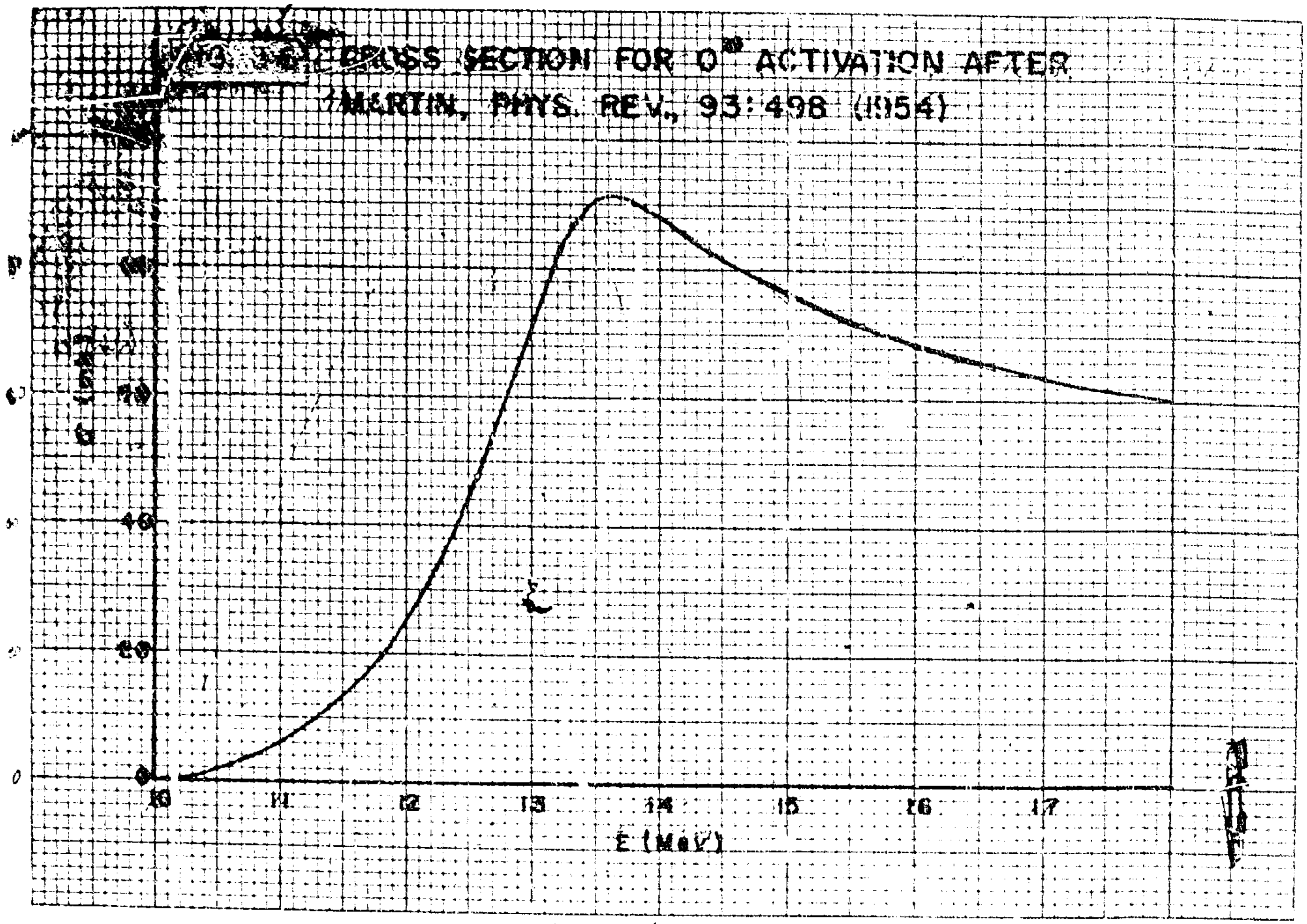

CROSS SECTION FOR O ACTIVATION AFTER
MARTIN, PHYS. REV., 93:498 (1954)
0
20
40
10
11
12
13
14
15
16
17
E (MeV)

FIG. 14-22

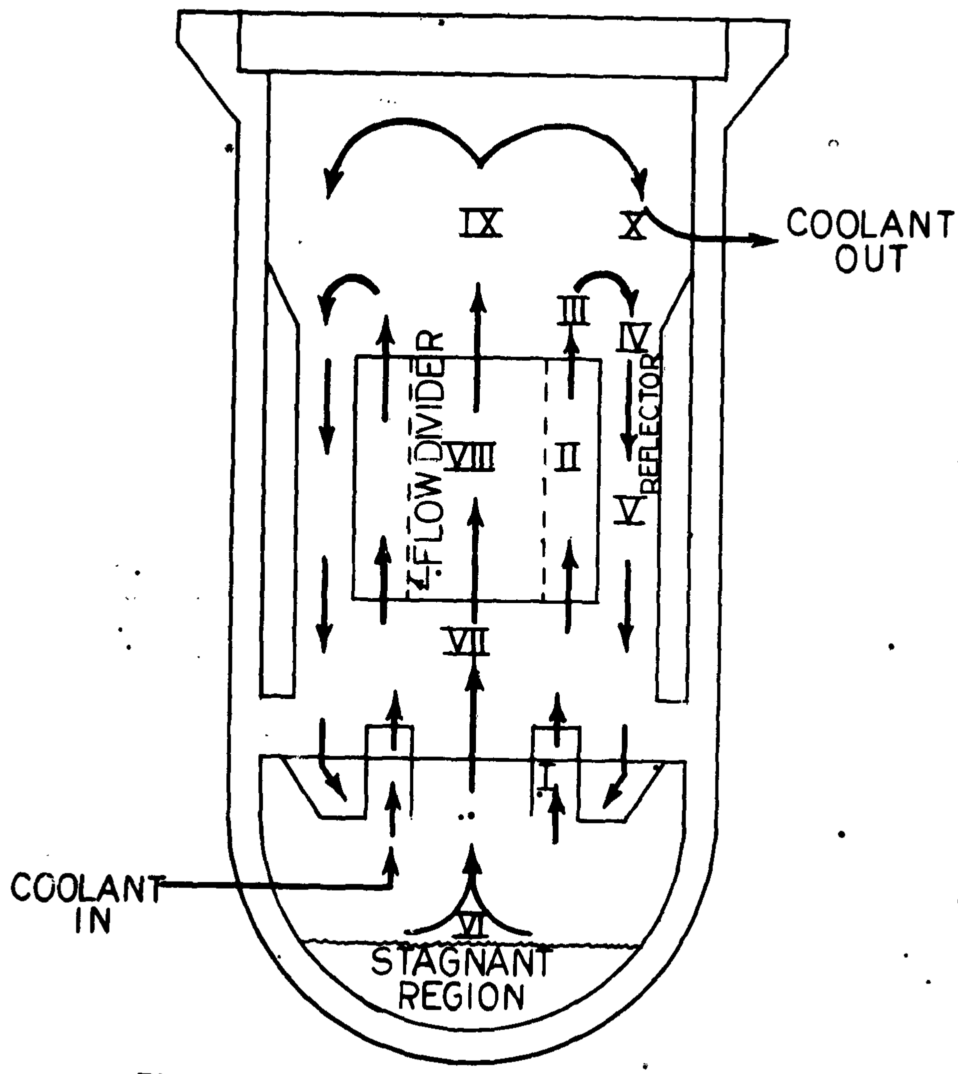


FIG. 14-23 SCHEMATIC OF ACTIVATING REGIONS OF TWO-PASS CORE.

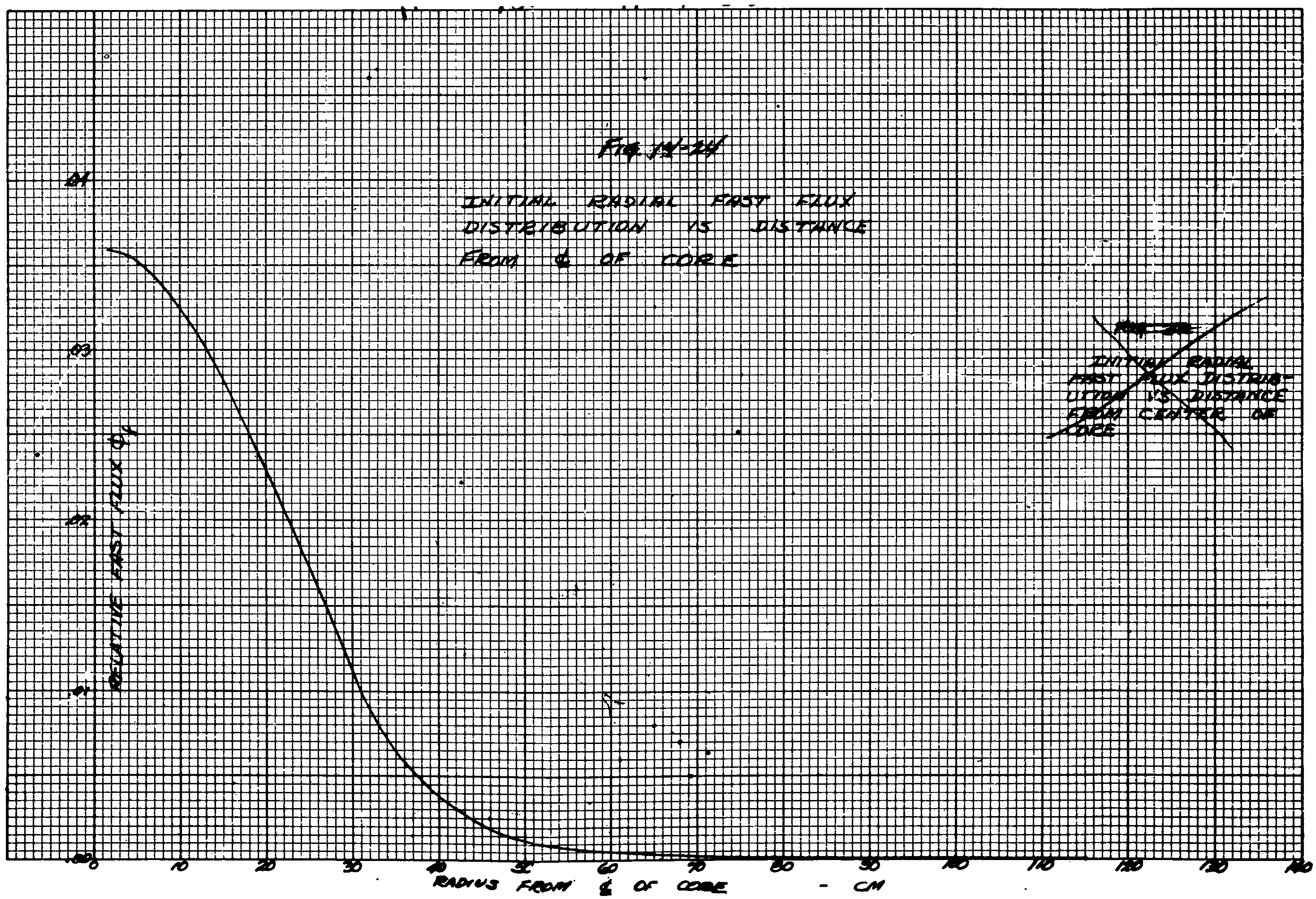


FIG. 14-24
INITIAL RADIAL FAST FLUX
DISTRIBUTION VS DISTANCE
FROM ℄ OF CORE
~~INITIAL RADIAL FAST FLUX DISTRIB-UTION VS DISTANCE FROM CENTER OF CORE~~
RELATIVE FAST FLUX ϕ_f
.04
.03
.02
.01
.000
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
RADIUS FROM ℄ OF CORE - CM

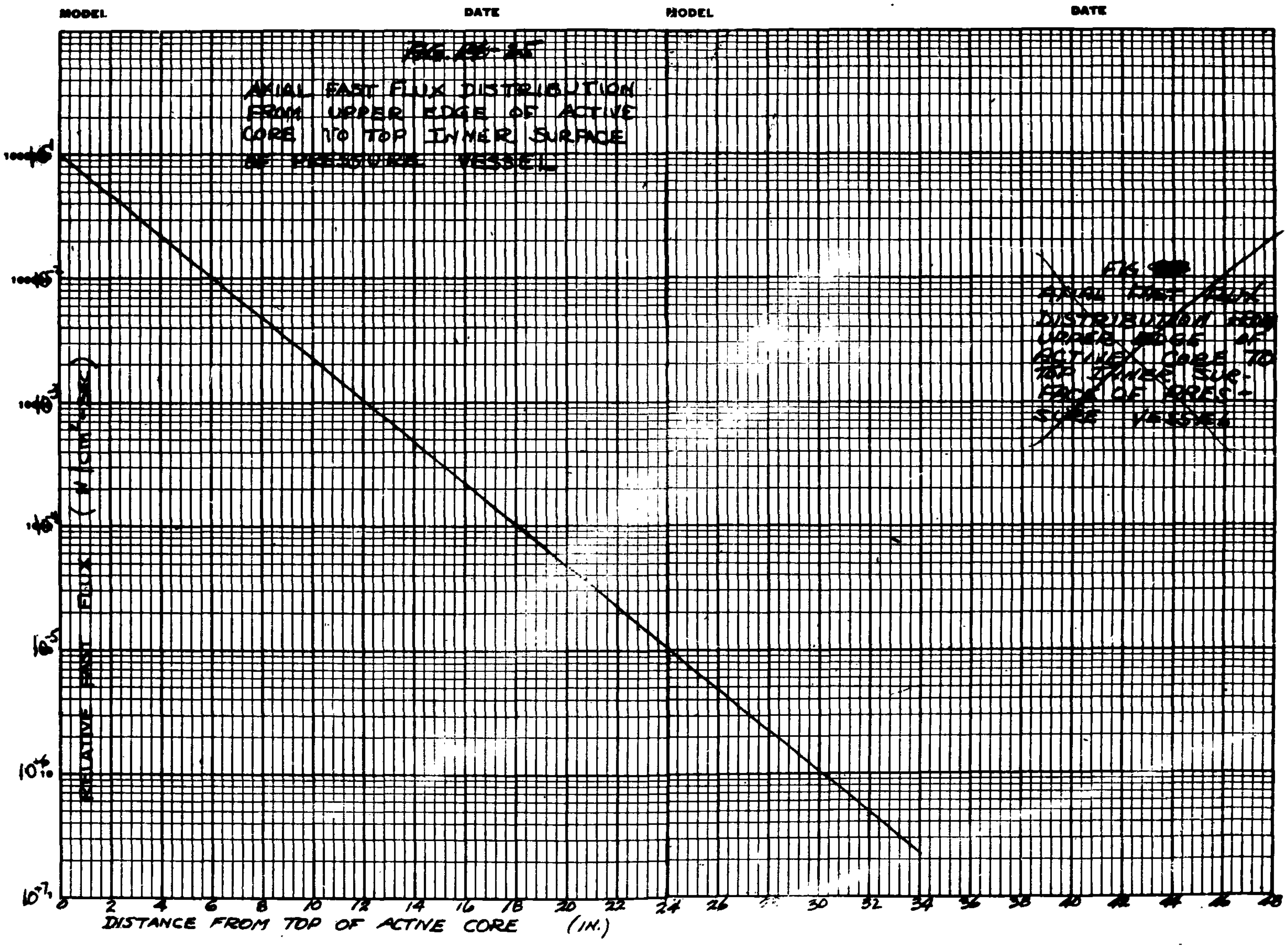

MODEL
DATE
MODEL
DATE
AXIAL FAST FLUX DISTRIBUTION
FROM UPPER EDGE OF ACTIVE
CORE TO TOP INNER SURFACE
OF PRESSURE VESSEL
RELATIVE FAST FLUX (#/CM²-SEC)
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30 32 34 36 38 40 42 44 46 48
DISTANCE FROM TOP OF ACTIVE CORE (IN.)

FIG. 14-26

INITIAL AXIAL FAST FLUX DISTRIBUTION

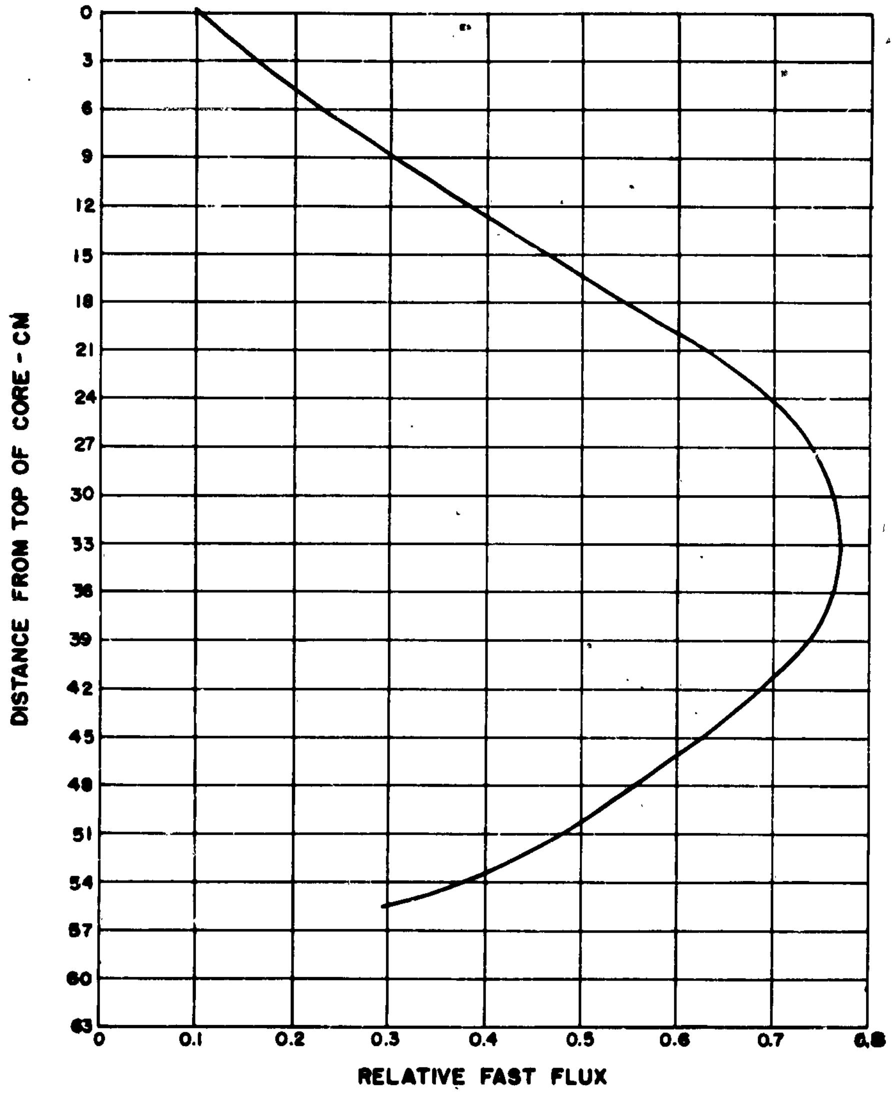
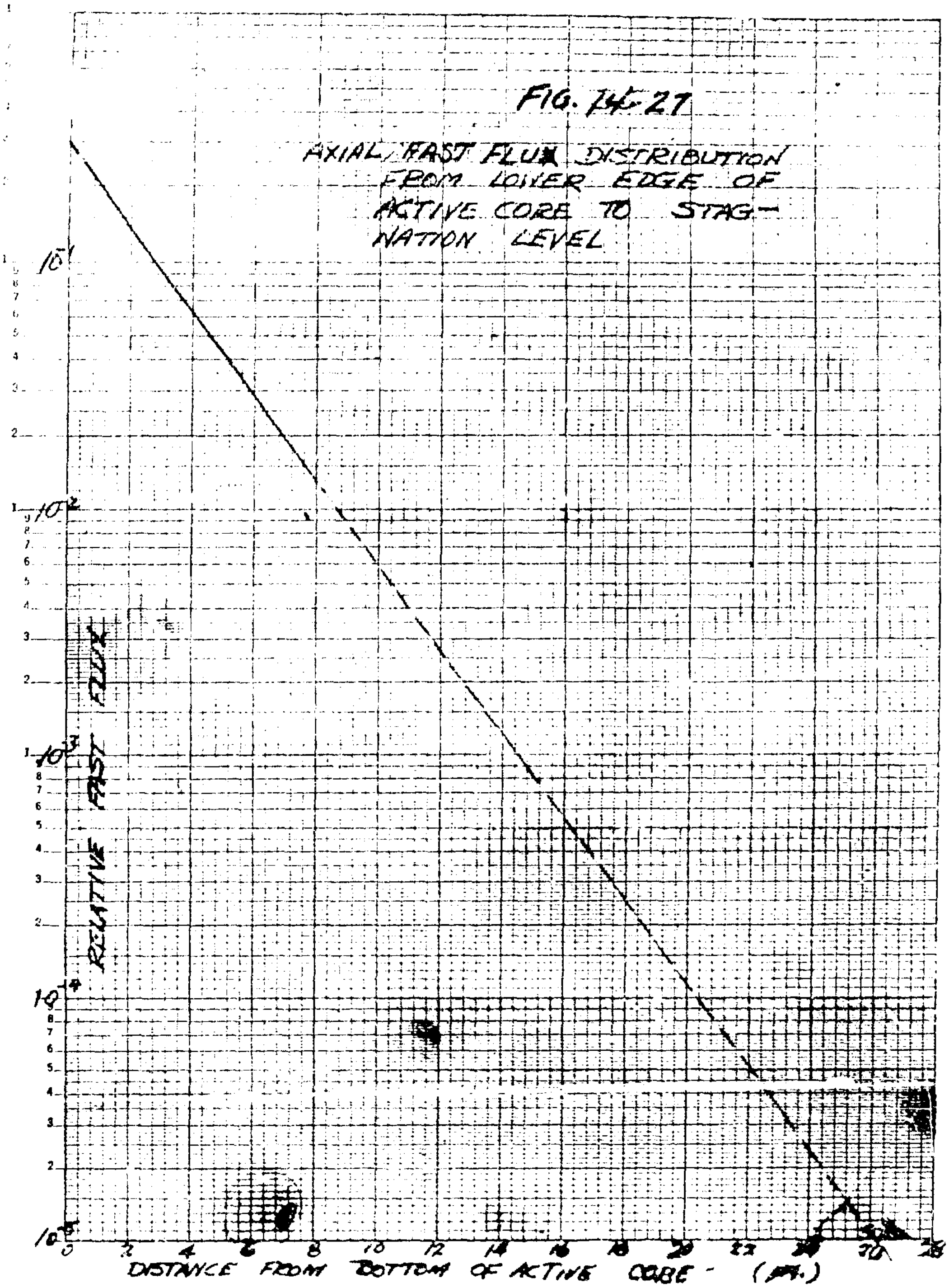


FIG. 14-27

AXIAL FAST FLUX DISTRIBUTION FROM LOWER EDGE OF ACTIVE CORE TO STAGNATION LEVEL



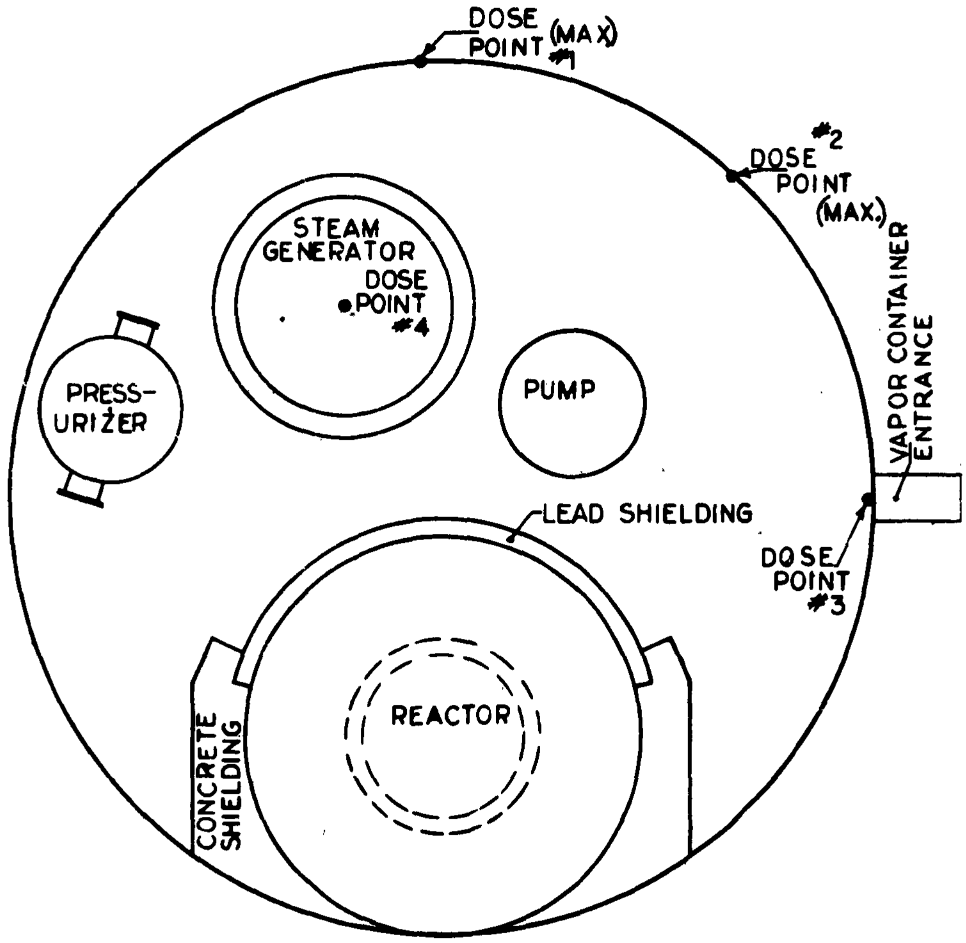


FIG. 14.28 SCHMATIC OF PLAN VIEW OF VAPOR CONTAINER SHOWING DOSE POINT POSITIONS USED IN RASH CODE.

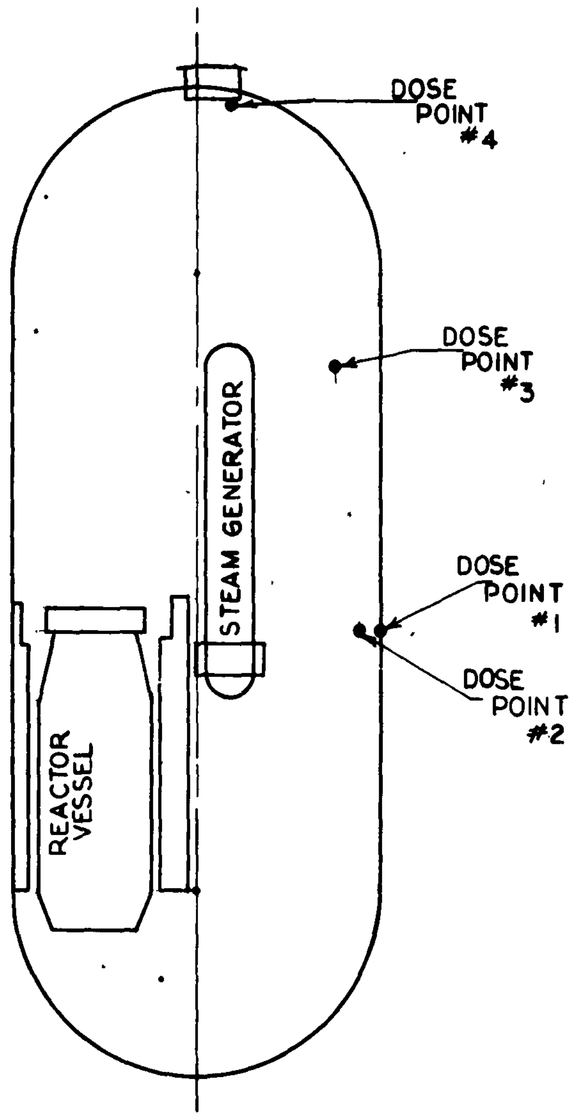
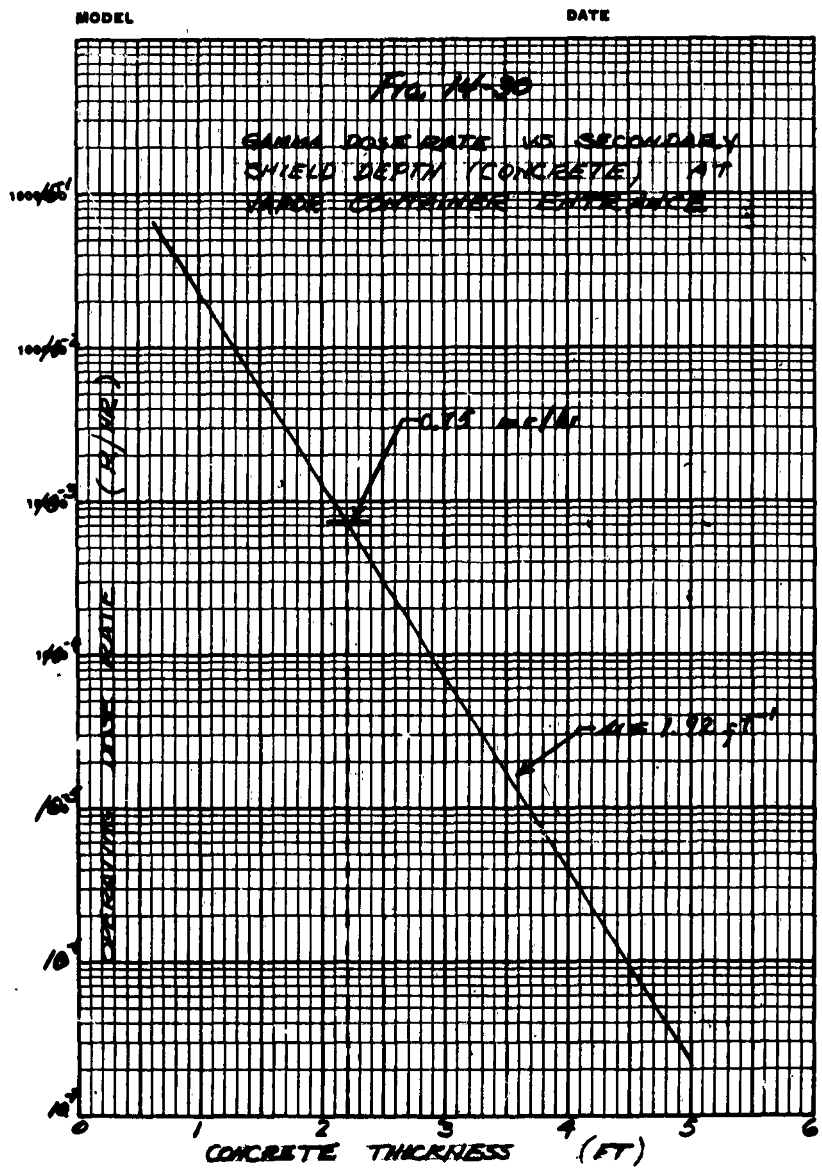


FIG. 14-29 SCHEMATIC OF ELEVATION VIEW OF VAPOR CONTAINER SHOWING DOSE POINT POSITIONS USED IN RAS-1 CODE.


MODEL
DATE
FIG. 14-30
GAMMA DOSE RATE VS SECONDARY SHIELD DEPTH (CONCRETE) AT VAPOR CONTAINER ENTRANCE
0.75 mr/hr
μ = 1.92 ft⁻¹
OPERATING DOSE RATE (R/HR)
0
1
2
3
4
5
6
CONCRETE THICKNESS (FT)

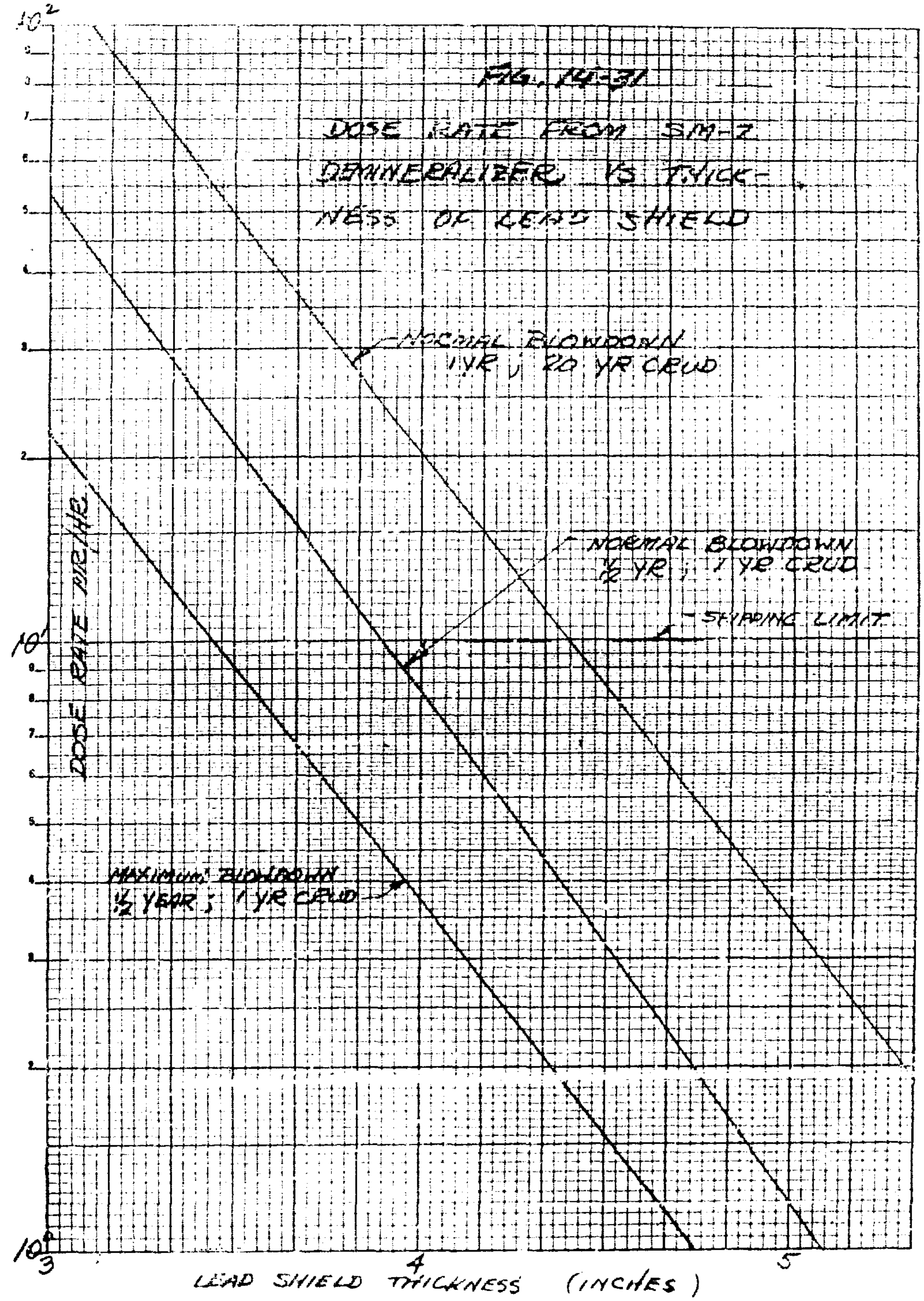
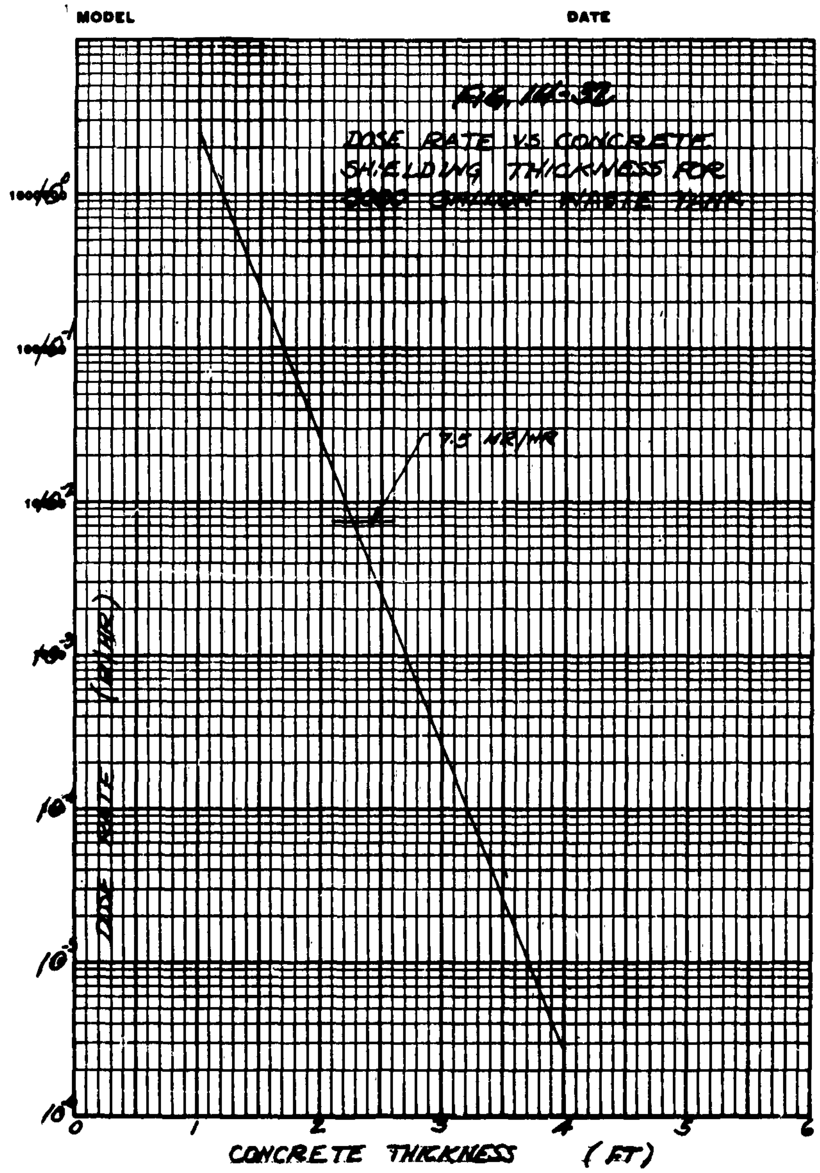
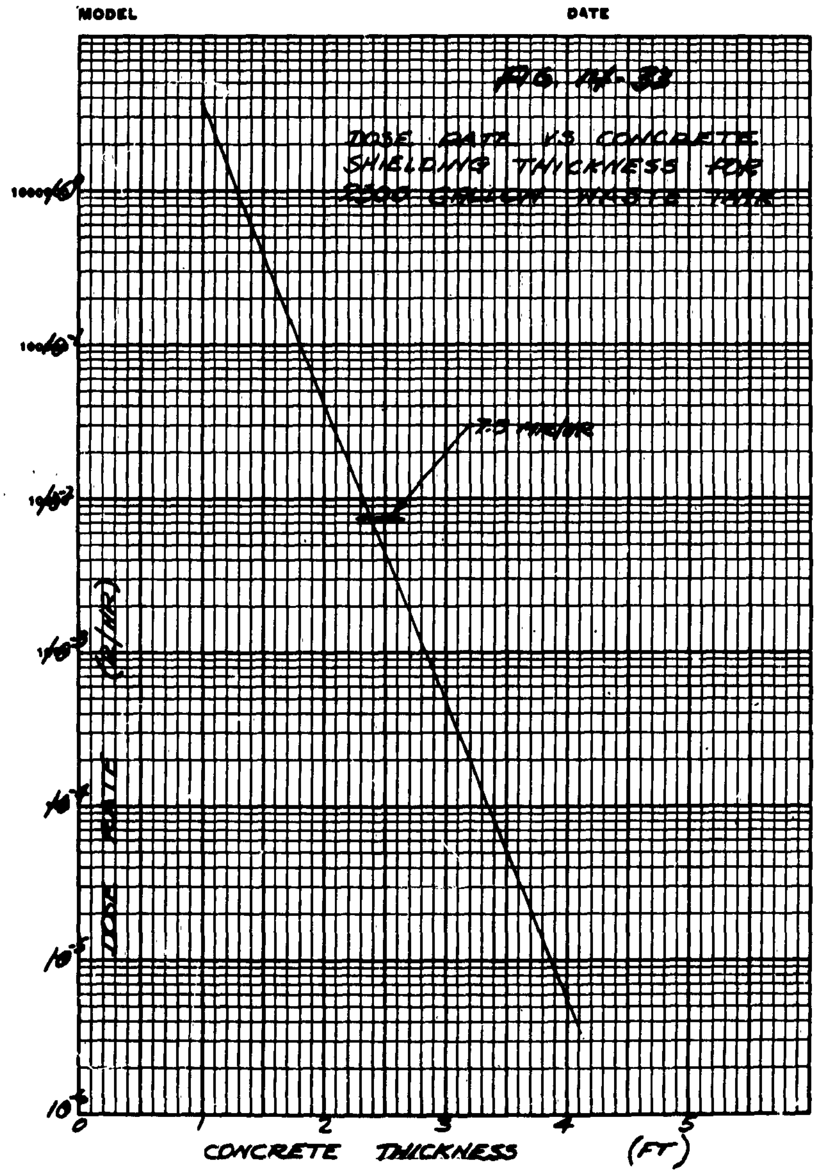


FIG. 14-31
DOSE RATE FROM SM-1 DEMINERALIZER VS THICKNESS OF LEAD SHIELD
NORMAL BLOWDOWN 1 YR ; 20 YR CRUD
NORMAL BLOWDOWN ½ YR ; 1 YR CRUD
SHIPPING LIMIT
MAXIMUM BLOWDOWN ½ YEAR ; 1 YR CRUD
DOSE RATE MR/HR
LEAD SHIELD THICKNESS (INCHES)
10^2
10^1
10^0
3
4
5



MODEL
DATE
DOSE RATE VS CONCRETE
SHIELDING THICKNESS FOR
7.5 MR/HR
DOSE RATE (R/HR)
CONCRETE THICKNESS (FT)


MODEL
DATE
FIG. IV-32
DOSE RATE VS CONCRETE SHIELDING THICKNESS FOR 2500 GALLON WASTE TANK
7.5 MR/HR
DOSE RATE (R/HR)
0
1
2
3
4
5
CONCRETE THICKNESS (FT)